Overview of Transient Nuclear Fuel Cycle Systems Studies at Los Alamos

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Topical Outline

(● indicates charts to be presented; remainder is backup)

- ➢ Summary of activities;
- ➢ Integrated NFC modeling;
- ➢ Simulation (NFCSim);
- ➢ Optimization:
  - - Nuclear Fuel Cycle (FCOPT);
  - - General Energy (US-MARKAL);
- ➢ Neutronics:
  - - Modeling support (High Pu/MA-recycle LWRs);
  - - Neutronics-based proliferation metrics;
- ➢ Yucca Mountain Business Model (YMBM);
- ➢ CEA/USDOE (ANL, LANL) Collaboration.

"Top-Level" Summary of FY03 Activities

- **Conduct CEA-LANL/ANL dynamic NFC model benchmarking and reference case studies (COSI-NFCSim):**
  - **- align processing, neutronics, costing, *etc.* databases;**
  - **- finalized NFC scenarios to be compared (France, US);**
  - **- investigate short- and long-term repository impacts (US), and long-term Pu inventory management strategies (France, US);**
- **Apply NFCSim *simulation model* development, in parallel with *optimization model* (FCOPT) to specific NFC scenarios, as suggested by (equilibrium) DELTA model;**
- **Advance fidelity of Yucca Mountain Business Model and integrate into optimization (FCOPT, MARKAL) and simulation (NFCSim) models;**
- **Initiate development of NFC optimization model in a broader (US) energy context (MARKAL).**

Integrated NFC Modeling

General Approach to Nuclear Fuel Cycle Analyses Used in AAA/AFCI Project: Specifics

- Scope scenario options/impacts using equilibrium (steady-state), "top-level" (aggregated processes) DELTA model:

  - Evaluate scenarios based on a range of performance indicators or metrics (*e.g*., cost, waste mitigation, proliferation risk, resource utilization);

  - Build scenarios based on coupled technologies presented in *multi-tiered* [LWROT/LWRMX(N)/FSB] configurations;

- Based on equilibrium analyses yielded by the DELTA model, perform *dynamic simulations* and *optimizations* on limited number of scenarios:

  - NFC Simulation Models: NFCSim (+ ORIGEN2.2);

  - NFC Optimization Model: FCOPT;

- Examine ANFC implications in a total energy context: MARKAL;

- SOTA neutronics (burn-up, depletion, reactivity, *etc.)* analysis support are crucial at all levels of ANFC modeling: ORIGEN2.2, Monteburns, MCNPX.

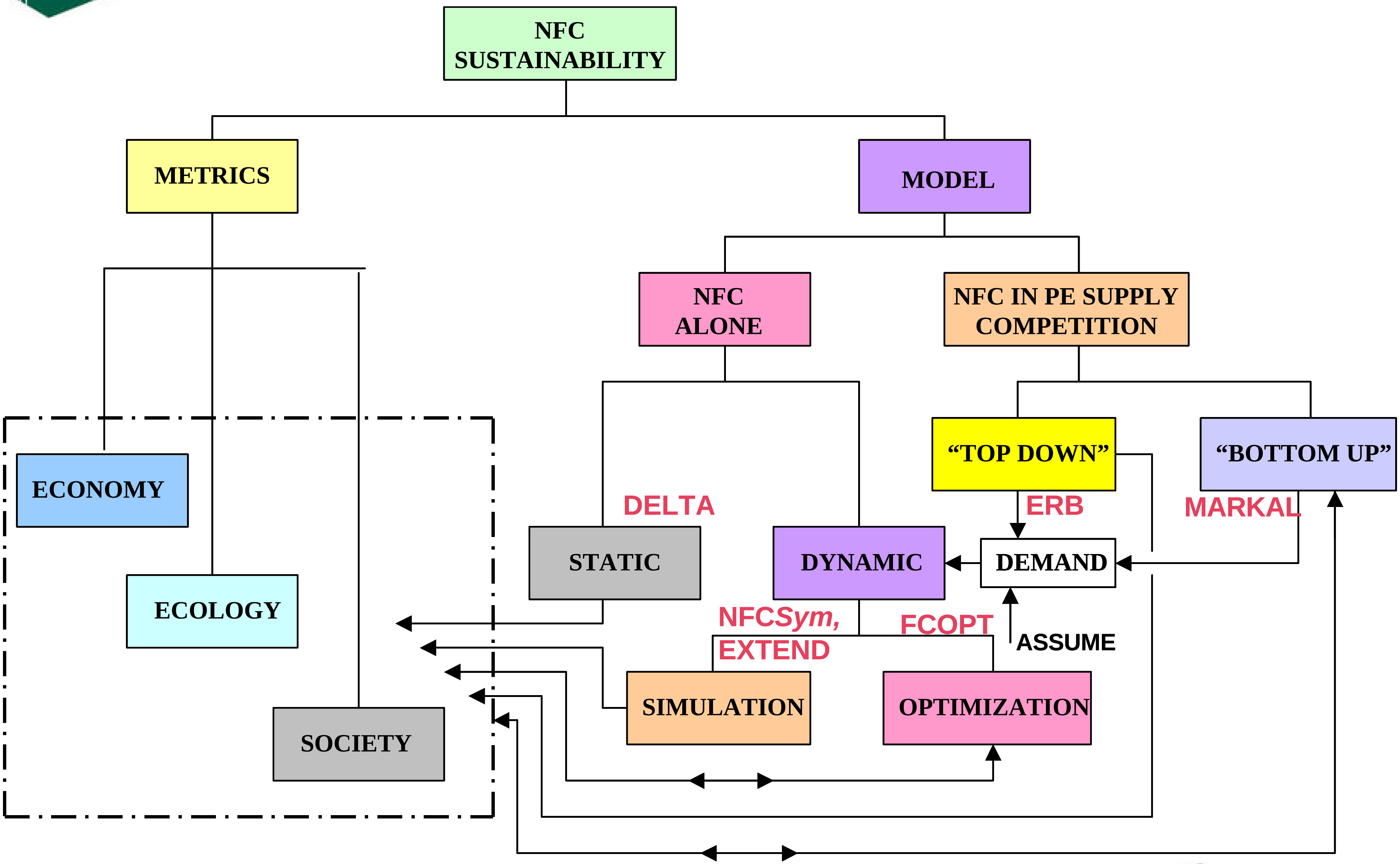
"Top-Level" Scenarios Suggested by CEA/DOE Collaboration for Time-Dependent Analyses

nsc	Tier-0		Tier-1		Tier-2
1	*lwrot*	SNF		~90/10 Pu/MA	FSB (ADS, FR*)
2	*lwrot*	SNF	*lwrmx(lr)* (2-3 recycle)	80/20 Pu/MA	FSB (ADS*, FR*)
3	*lwrot*	SNF	*lwrmx(hr)* (CORAIL, 8-10 recycle)	~1/99 Pu/MA	FSB (ADS*, FR)
4	*lwrot*	SNF	*htgr* (deep burn, 1 recycle)	~10/90 Pu/MA	FSB (ADS*, FR)

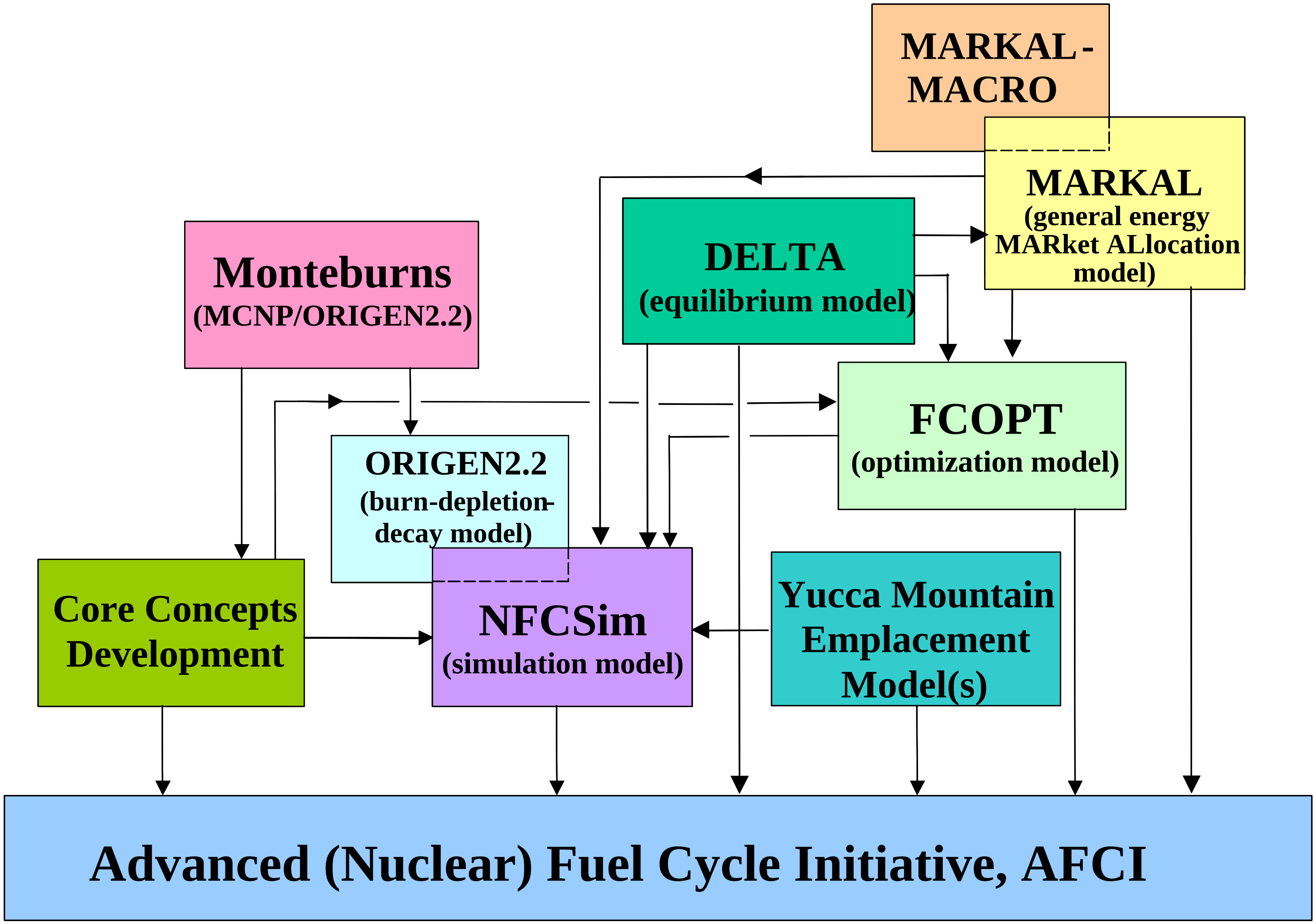
lwrot	once-through LWR
lwrmx(lr)	low-recycle MOXed LWR
lwrmx(hr)	high-recyle MOXed LWR
FSB	fast spectrum burner
ADS	accelerator-driven system
FR	fast (critical) reactor
nsc	scenario grouping
MA	minor actinide
SNF	spent nuclear fuel (*lwrot*)

* Preferred on the basis of equilibrium economics, except for *nsc* = 2, where within uncertainties both are equivalent.

ANFC Modeling Relationships, Scope, and Options and Approaches Being Pursued at Los Alamos



An Integrated Approach to AFCI Modeling as Pursued at Los Alamos



Simulations (NFCSim)

NFCSim Models the Flow of Nuclear Materials Through the Nuclear Fuel Cycle

- **NFCSim tracks mass flow at the level of discrete reactor fuel charges/discharges for the US, logging in time the following:**
  - isotopic distribution;
  - originating reactor;
  - arrival, departures, and irradiation dates.
- **Processes/facilities modeled include:**
  - mining & milling,
  - conversion,
  - enrichment,
  - fuel fabrication,
  - reactor,
  - onsite storage,
  - interim storage,
  - separations,
  - transportation,
  - repository.

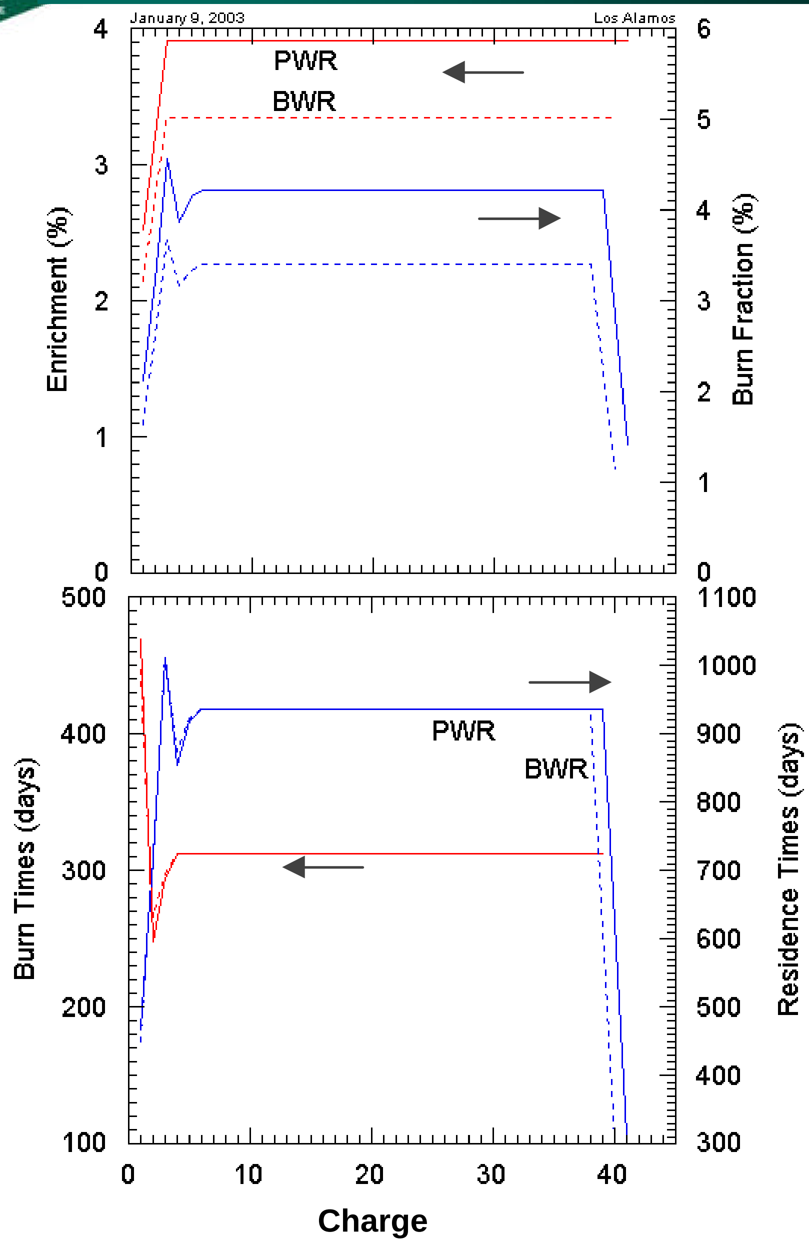
NFCSim Models the Flow of Nuclear Materials Through the Nuclear Fuel Cycle (cont.-1)

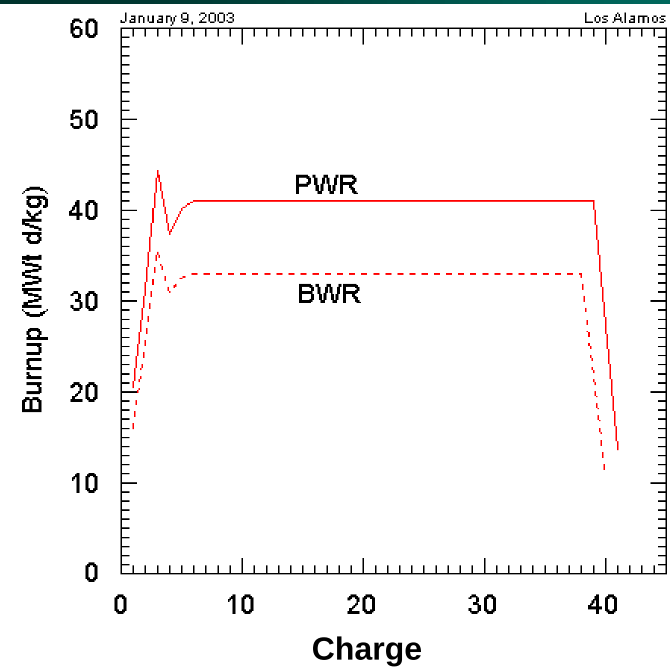
- **Simulation is event driven:**
  - **simulation proceeds sequentially from event to event;**
  - **event durations are input as integers with the specified units ranging from years to seconds (*e.g*., 18 months instead of 1.5 years) and translated internally into milliseconds.**
- **Simulation begins with present-day US fleet of commercial nuclear reactors (IAEA, EIA):**
  - **PWRs;**
  - **BWRs.**
- **Residence times of isotopes of interest are recorded for eventual use in proliferation-resistance model;**
- **Costs are tracked using a methodology similar to that used in the the DELTA(equilibrium) and FCOPT(optimization) models:**
  - **system-wide Cost of Electricity;**
  - **discounted Life-Cycle Cost (LCC), a new feature.**

NFCSim Models the Flow of Nuclear Materials Through the Nuclear Fuel Cycle (cont.-2)

- **Neutronics (burn-up, depletion, *etc*.) are provided by a directly coupled ORIGEN2.2 model that uses (recycle-dependent) cross sections updated by separate Monteburns computation. Allows analysis of:**
  - **non-equilibrium nature of fuel cycle (*i.e.,* beginning- and end-of-life transients);**
  - **multiple recycles of Pu and/or MA;**
  - **activity/radiotoxicity;**
  - **heat load.**

Startup and Shutdown Transients are Modeled *per Charge* for Each Reactor





Assumptions Used in NFCSim Example

- Simulation starts with the US commercial nuclear fleet;
- Reactor availability begins at 85%;
- Plant life is assumed to be 40 years, unless an extension has been granted, is being reviewed, or will be requested;
- Burn is 40 MWt d/kg for existing reactors and 55 MWt d/kg for new reactors;
- SNF must be 7 years old before it can be moved from cooling storage.

Assumptions Used in NFCSim Example (cont.-1)

- **Projected Yucca Mountain schedule is is used:**
  - **A total of 4,300 shipments (*i.e.*, a shipment is a discharge);**
  - **Shipments begin in 1/4/2010;**
  - **Shipments to Yucca Mountain occur over next 24 years;**
  - **In full operation, 200 shipments per year are assumed;**
  - **Assume shipping activity ramps up over 4 years:**
    - **instantaneous number of shipments during ramp period;**

$$N(t) = 300\left\{1 + \sin\left(\frac{p}{2}\left[\frac{t}{1458} - \frac{1}{2}\right]\right)\right\}$$

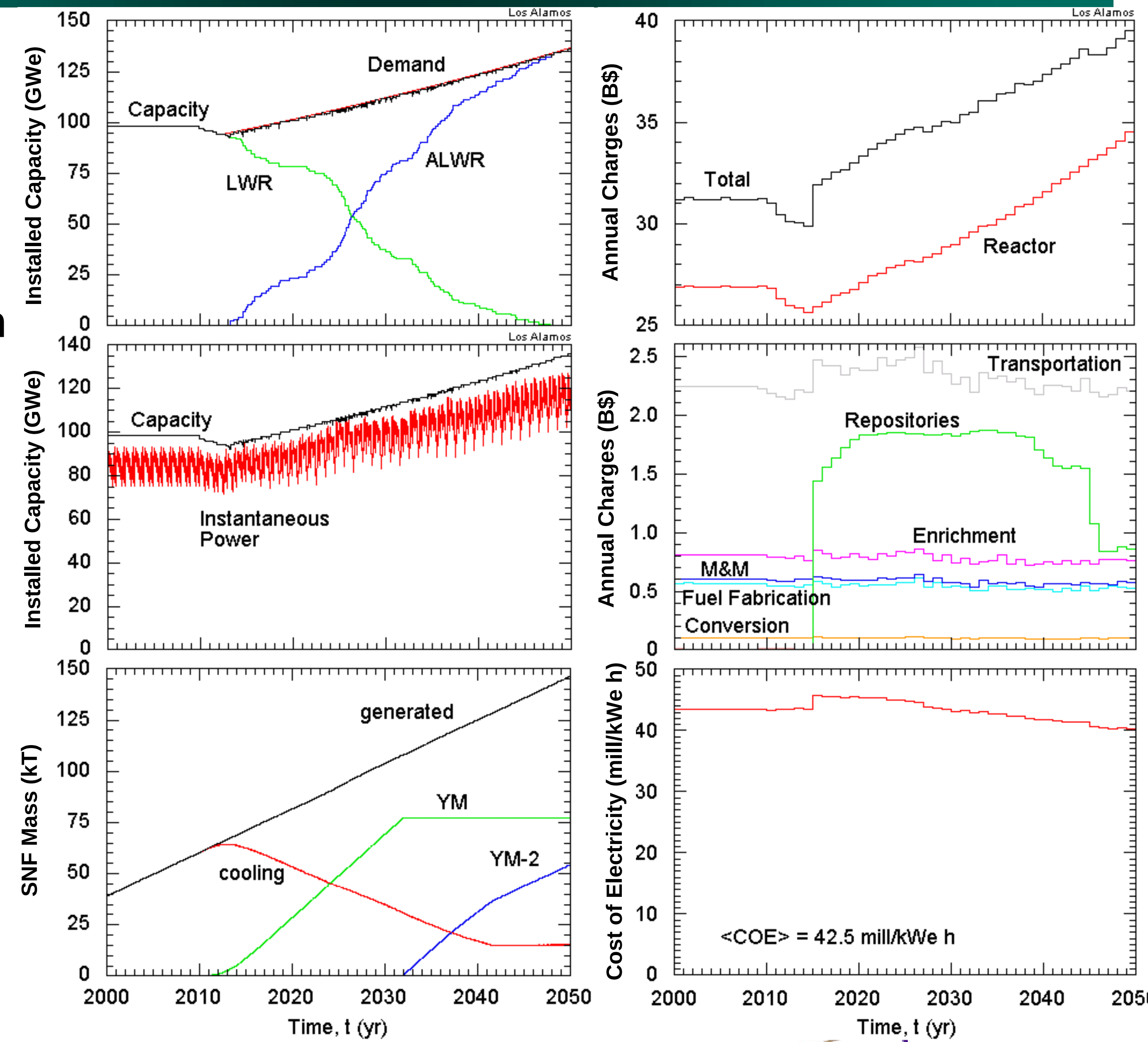
    - **cumulative number of shipments during ramp period.**

$$C(t) = 300\left\{\frac{t}{1458} - \cos\left(\frac{p}{2}\left[\frac{t}{1458} - \frac{1}{2}\right]\right)\right\}$$

Sample NFCSim Result: Nuclear Resurgence Scenario Based on ALWRs

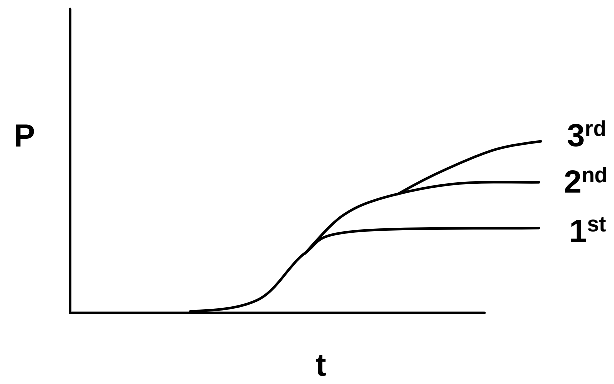
- Existing fleet of nuclear reactors supplies electricity until end of third quarter, 2012, when demand starts increasing 1% per year.

- Repository opens January 4, 2010.

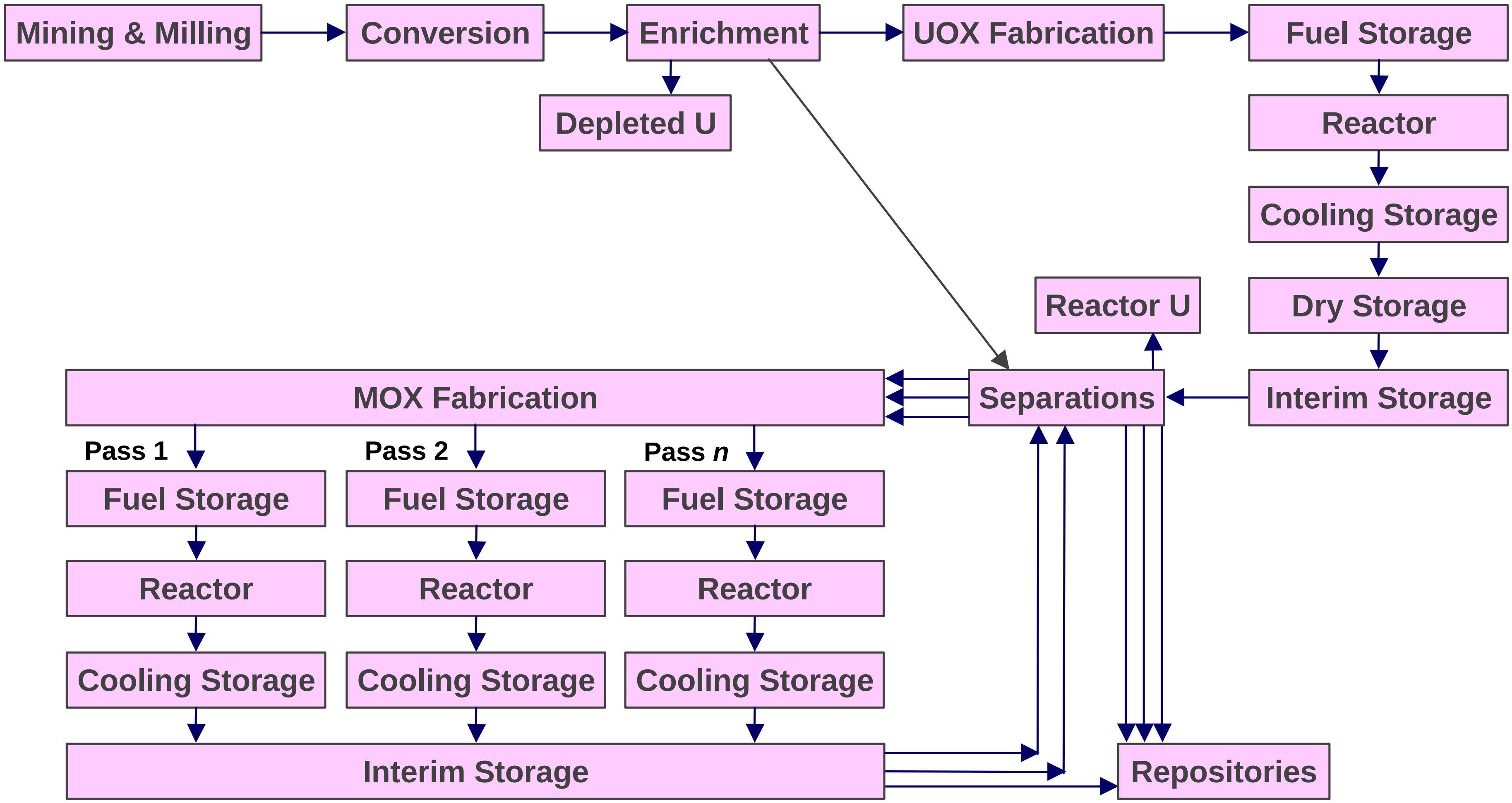


Implementation Plan for Series-I Simulation

- Current LWR fleet runs to shutdown;
- ALWRs replace LWRs per onset of demand at t_1;
- Repository opens at t_2;
- Reprocessing starts at half capacity at t_3;
- MOX fuel fabrication starts at half capacity at t_4;
- Burn MOX in ALWRs:
  - first-pass starts at t_5;
  - second-pass starts at t_6;
  - third-pass starts at t_7;
  - fourth-pass starts at t_8.
- Staged increases in:
  - burnup;
  - availability.

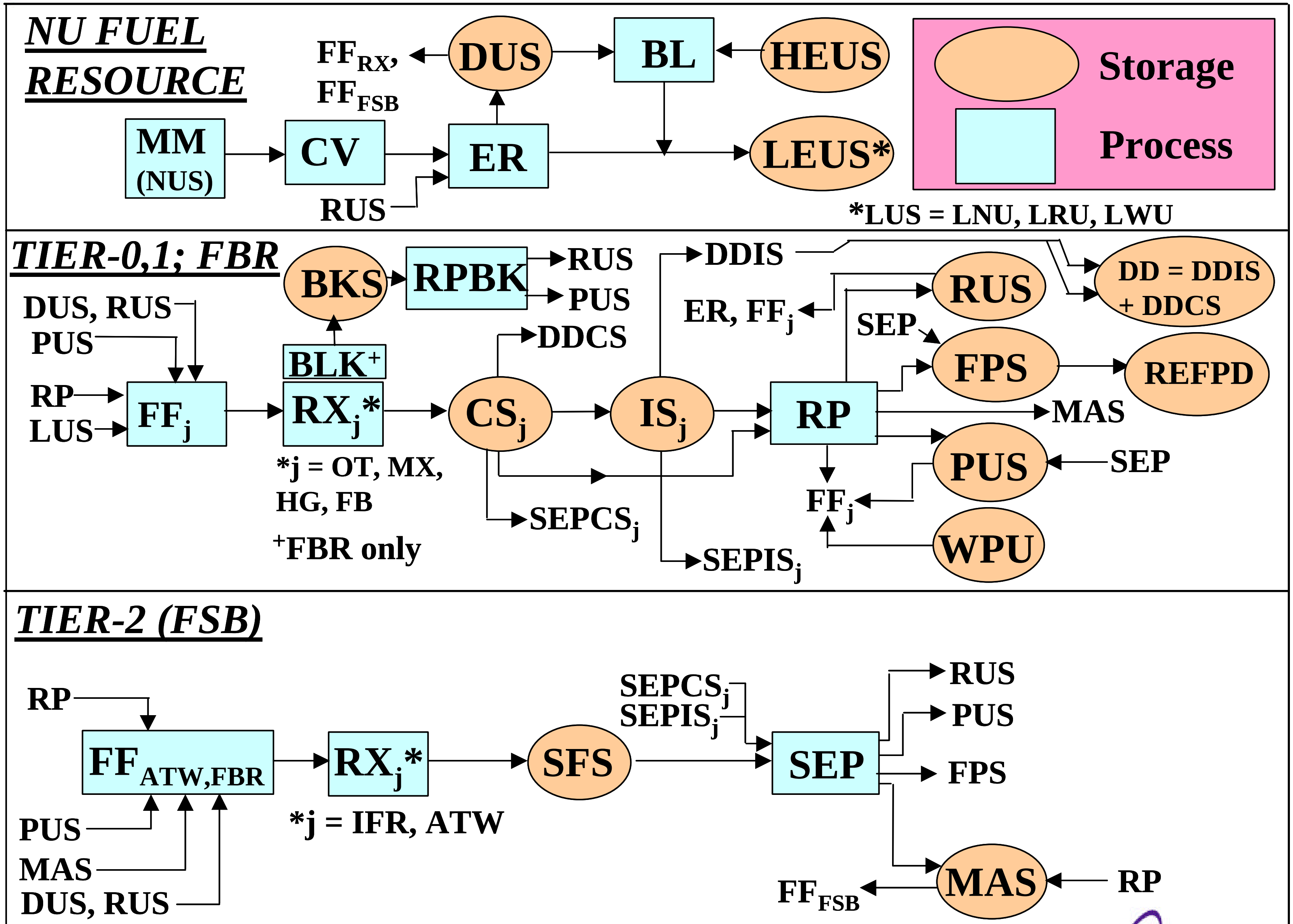


A Schematic Depicting Flow of Charges in NFCSim for Series 1

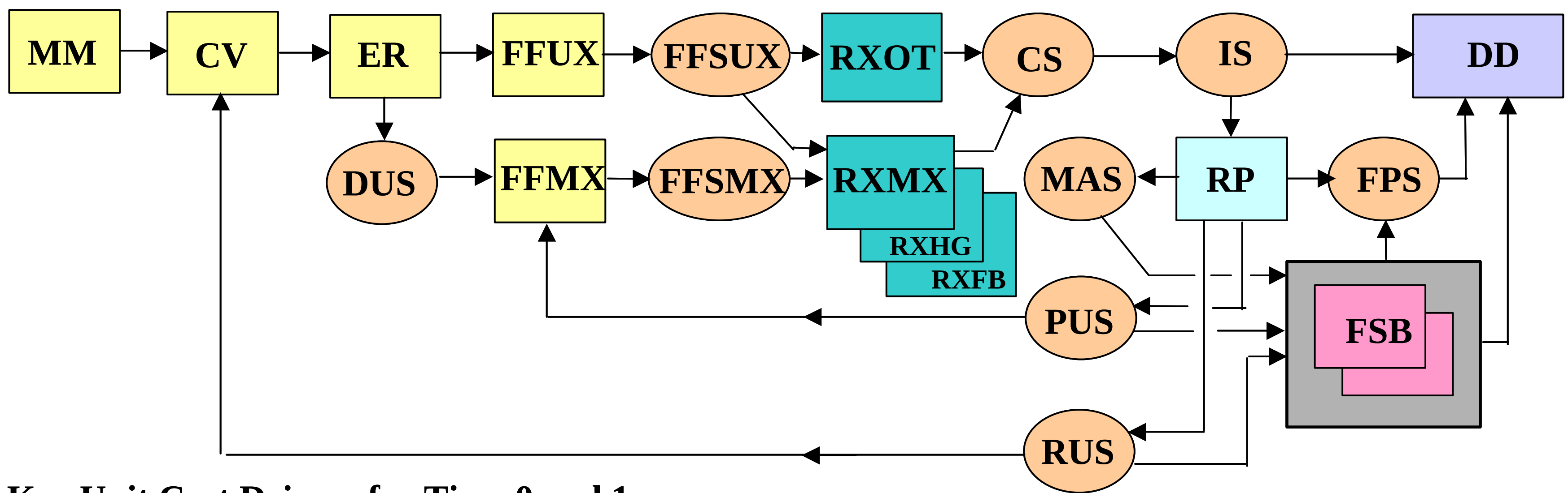


Optimization: Nuclear Fuel Cycle (FCOPT)

Overall Mass Flows in NFC Optimization Model FCOPT



Tier-0,1,2 Mass Flows in NFC Optimization Model FCOPT

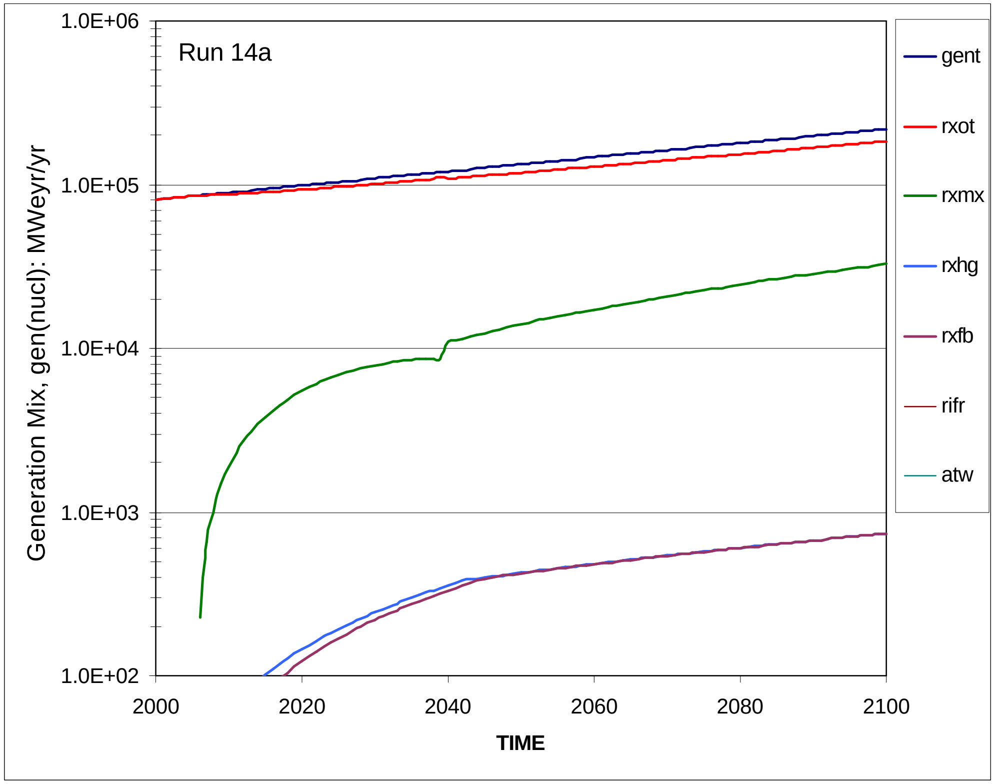


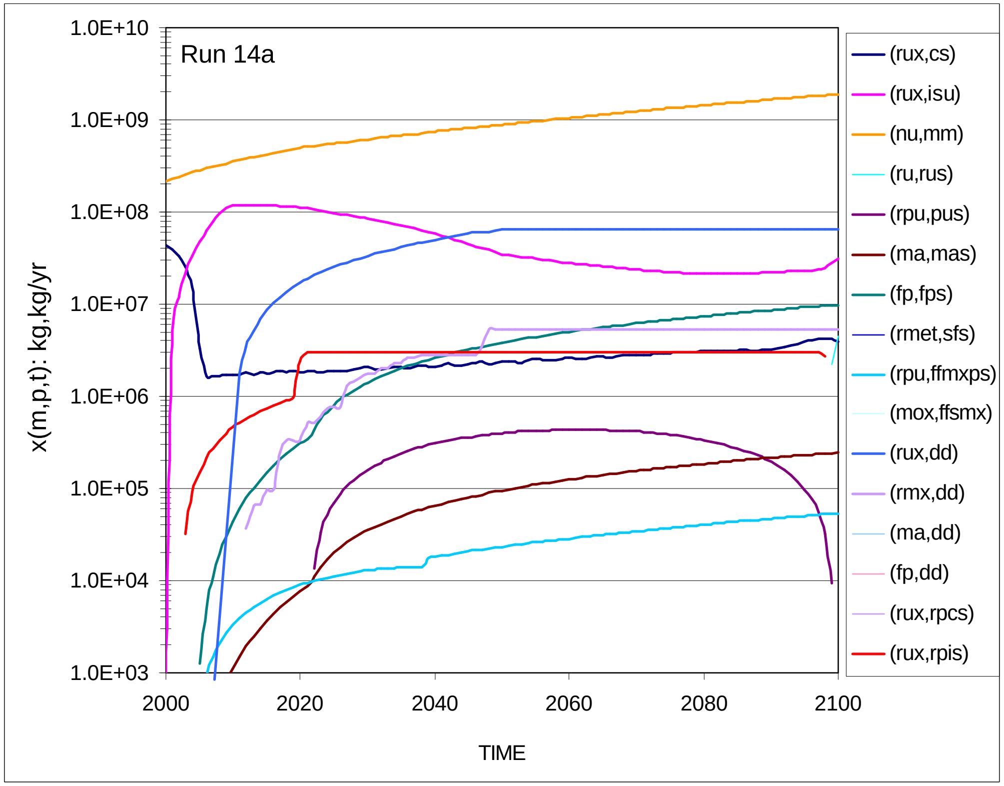
Key Unit Cost Drivers for Tiers 0 and 1

UCMM UCFF UCFFS UTC {UCCS, UCIS} UCPUS UCRP UCDD

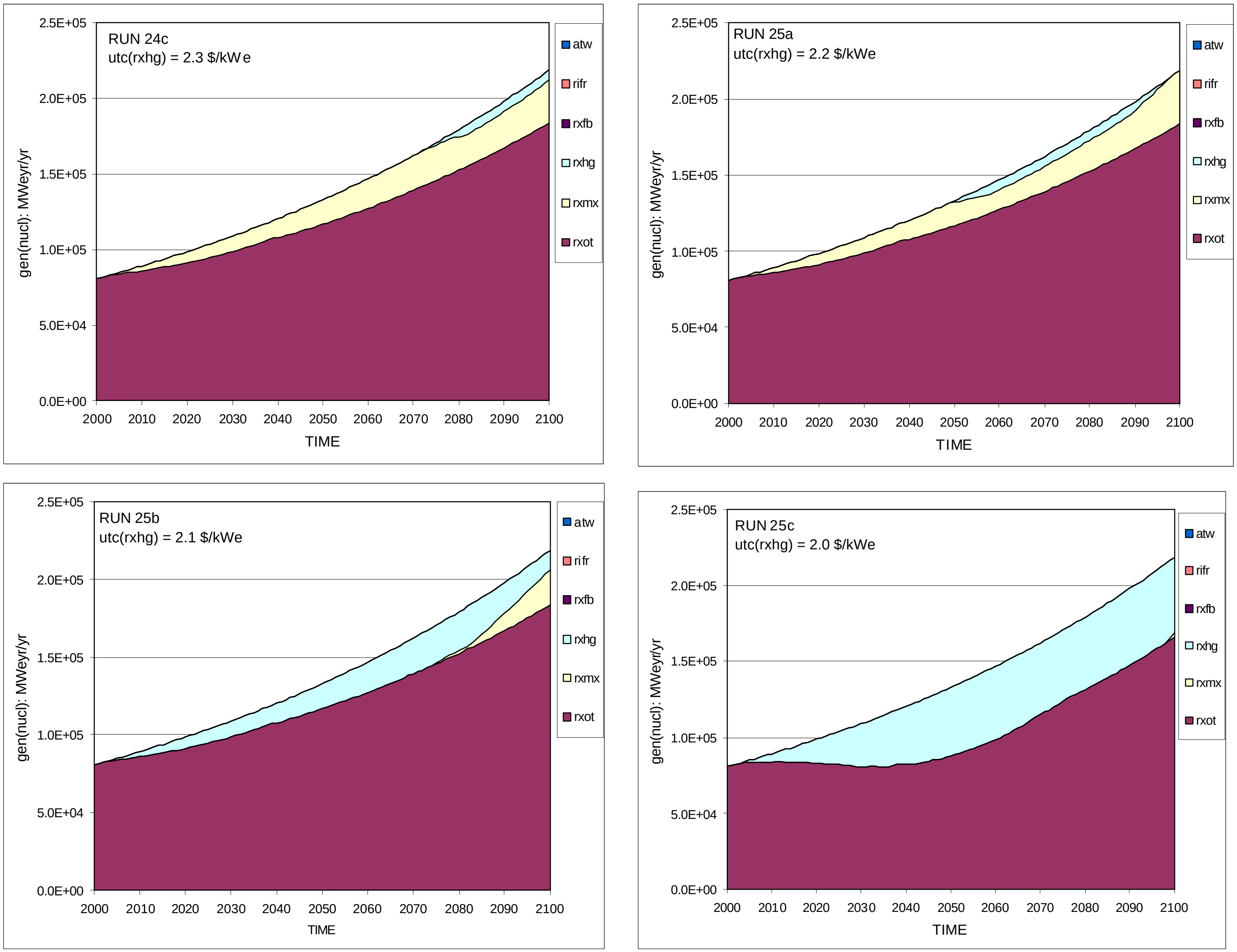
MM	Mining and Milling	DUS	Depleted Uranium Storage
CV	Conversion	RUS	Reactor Uranium Storage
ER	Enrichment	PUS	Plutonium Storage
FF	Fuel Fabrication	MAS	Minor Actinide Storage
UX	Uranium Oxide	FPS	Fission Product Storage
MX	Mixed Oxide	DD	Direct Disposal
FFS	Fresh Fuel Storage	FSB	Fast-Spectrum Burner (Tier-2 Systems)
RX	Reactor Technology	HG	Gas-Cooled RX
RP	Reprocessing Technology	FB	Fast-Breeder RX

Example FCOPT Result: Time Evolution of Generation Mix and Material Flows and Inventories

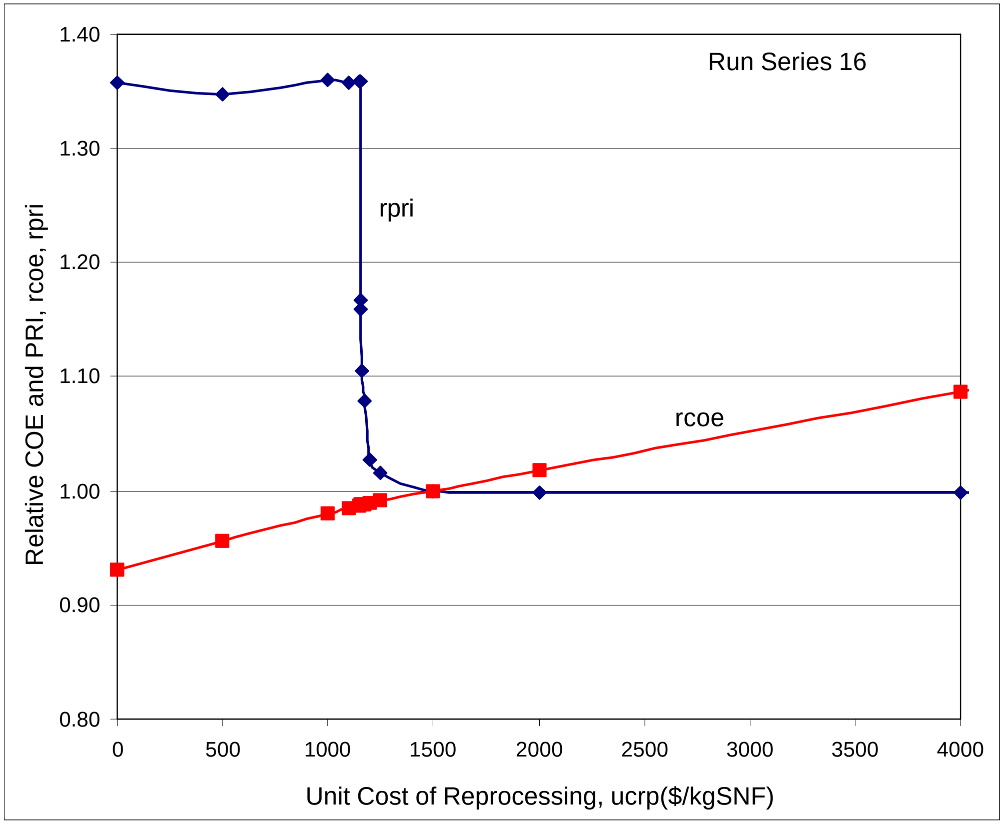


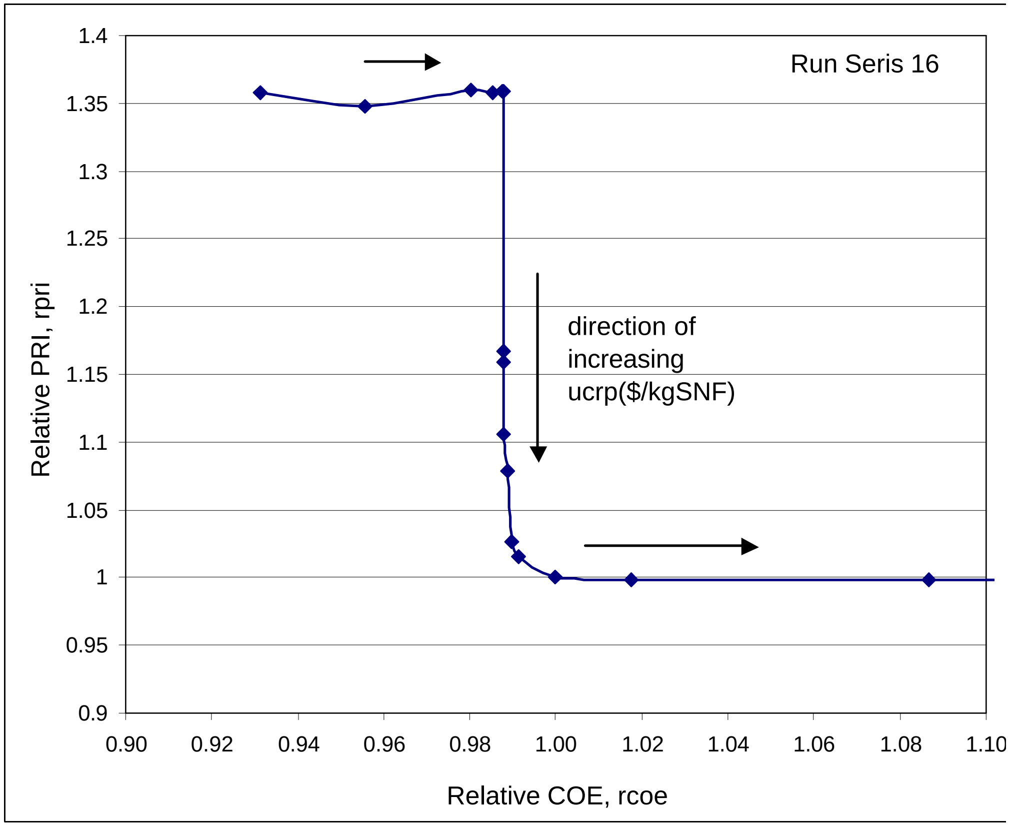


Example FCOPT Result: Time-dependent Generation Mix for a Range of HTGR Unit Total Costs, *utc($/We)*



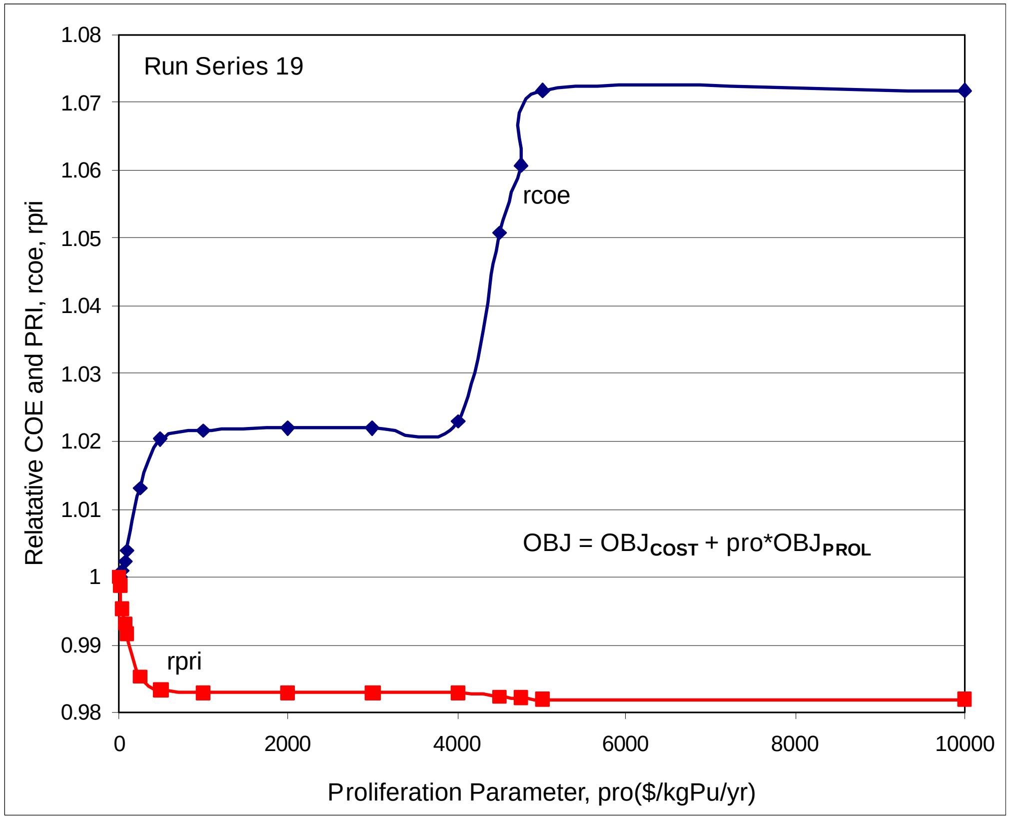
Example FCOPT Result: Dependence of Relative Cost, *rcoe*, and Proliferation Risk Index, *rpri*, on Reprocessing Unit Cost, *ucrp($/kgSNF)*

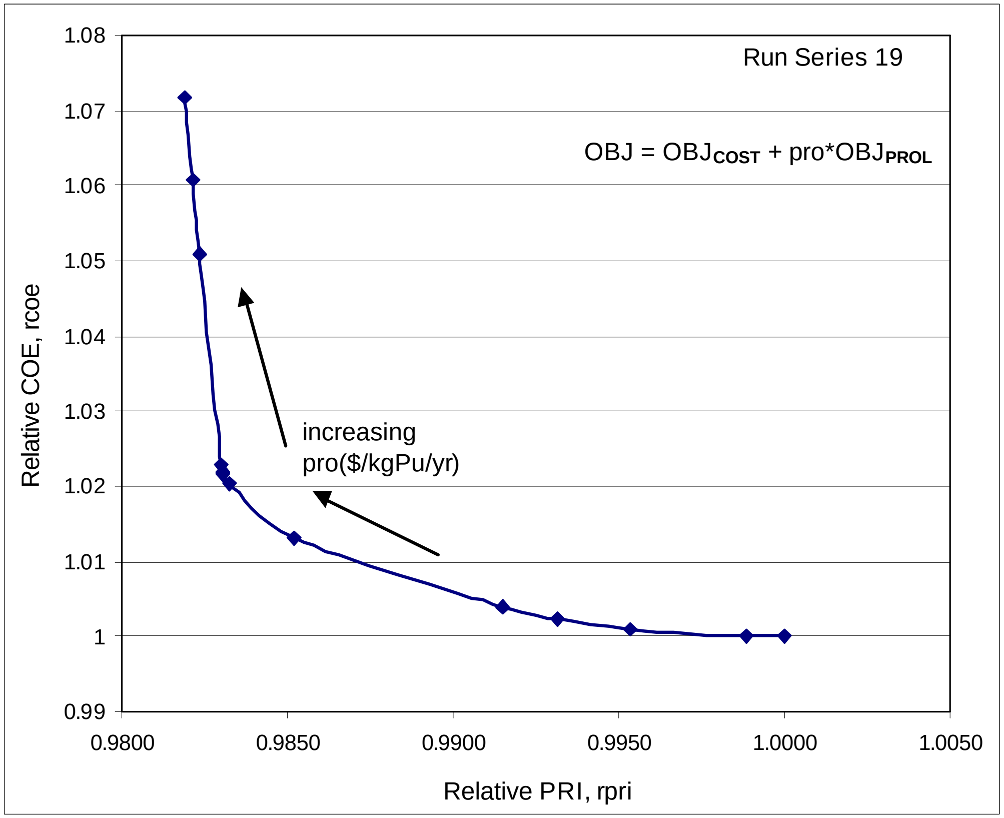




Note: Increased cost of reprocessing both increases the overall cost of electricity and decreases the amount of plutonium having high proliferation attractiveness level; left frame is a direct comparison plot, and right frame is a cost-risk correlation plot.

Example FCOPT Result: Dependence of the Relative Cost and Relative Proliferation Risk, *rcoe* and *rpri*, on the Cost-Proliferation Coupling Coefficient, *pro*



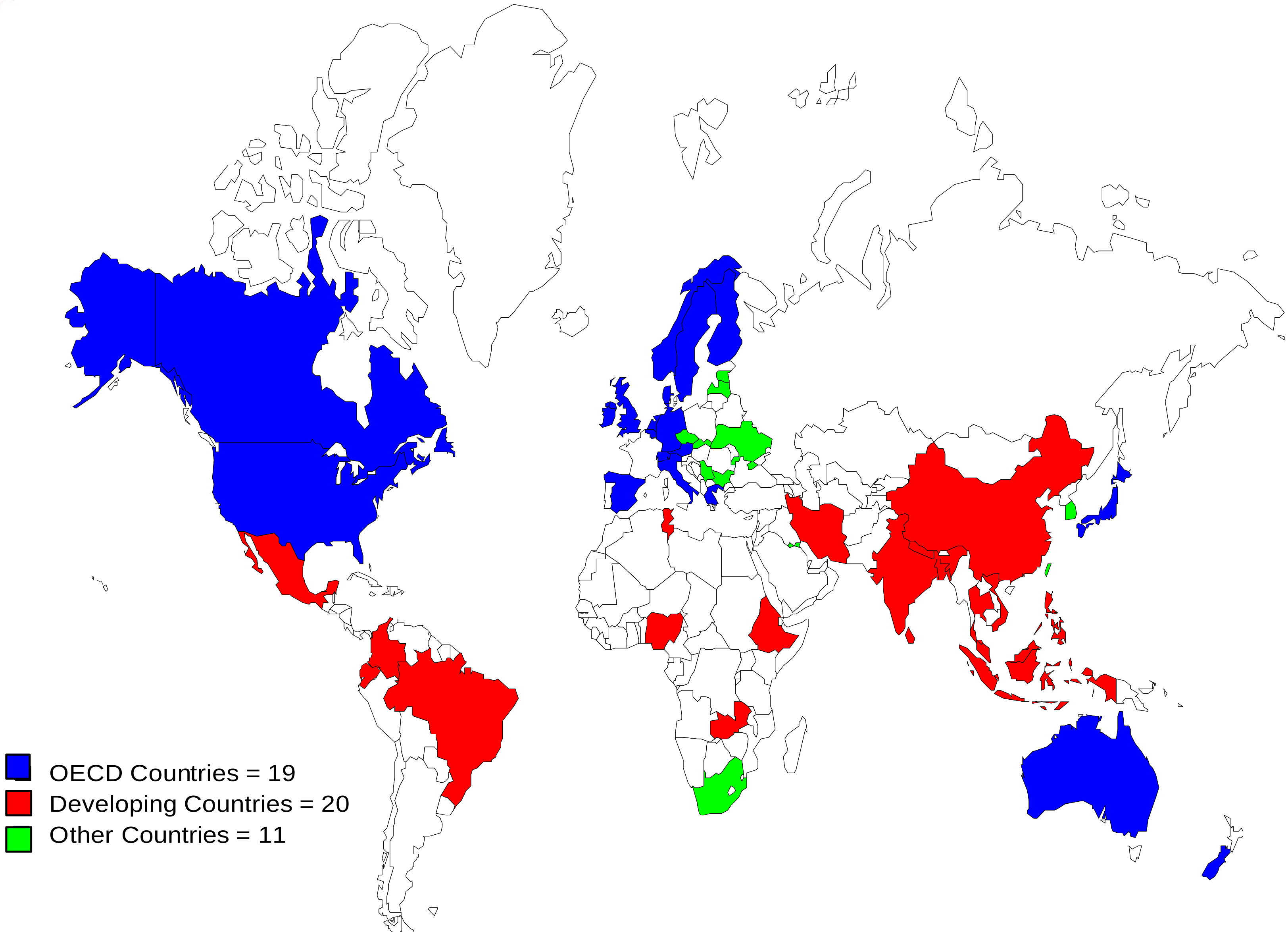


Optimization: General Energy (US-MARKAL)

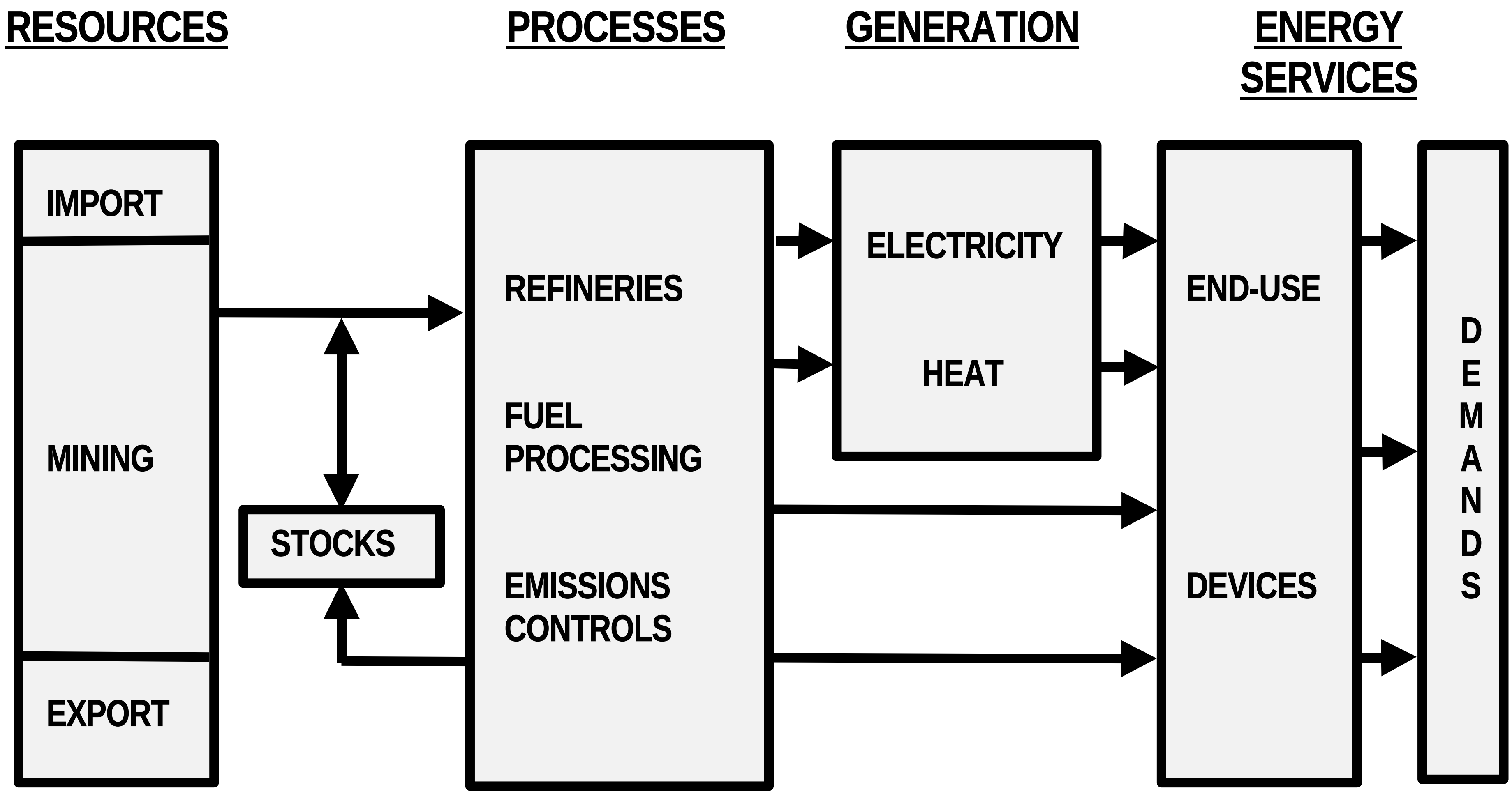
What MARKAL Does

- Identifies least-cost solutions for energy system planning.
- Evaluates options within the context of the entire energy/materials system by:
  - balancing all supply/demand requirements;
  - ensuring proper process/operation;
  - monitoring capital stock turnover;
  - adhering to environmental & policy restrictions.
- Selects technologies based on life-cycle costs of competing alternatives;
- Establishes baselines and the implications of alternate futures;
- Provides estimates of:
  - energy/material prices;
  - demand activity;
  - technology and fuel mixes;
  - GHG and other emission levels;
  - mitigation and control costs.

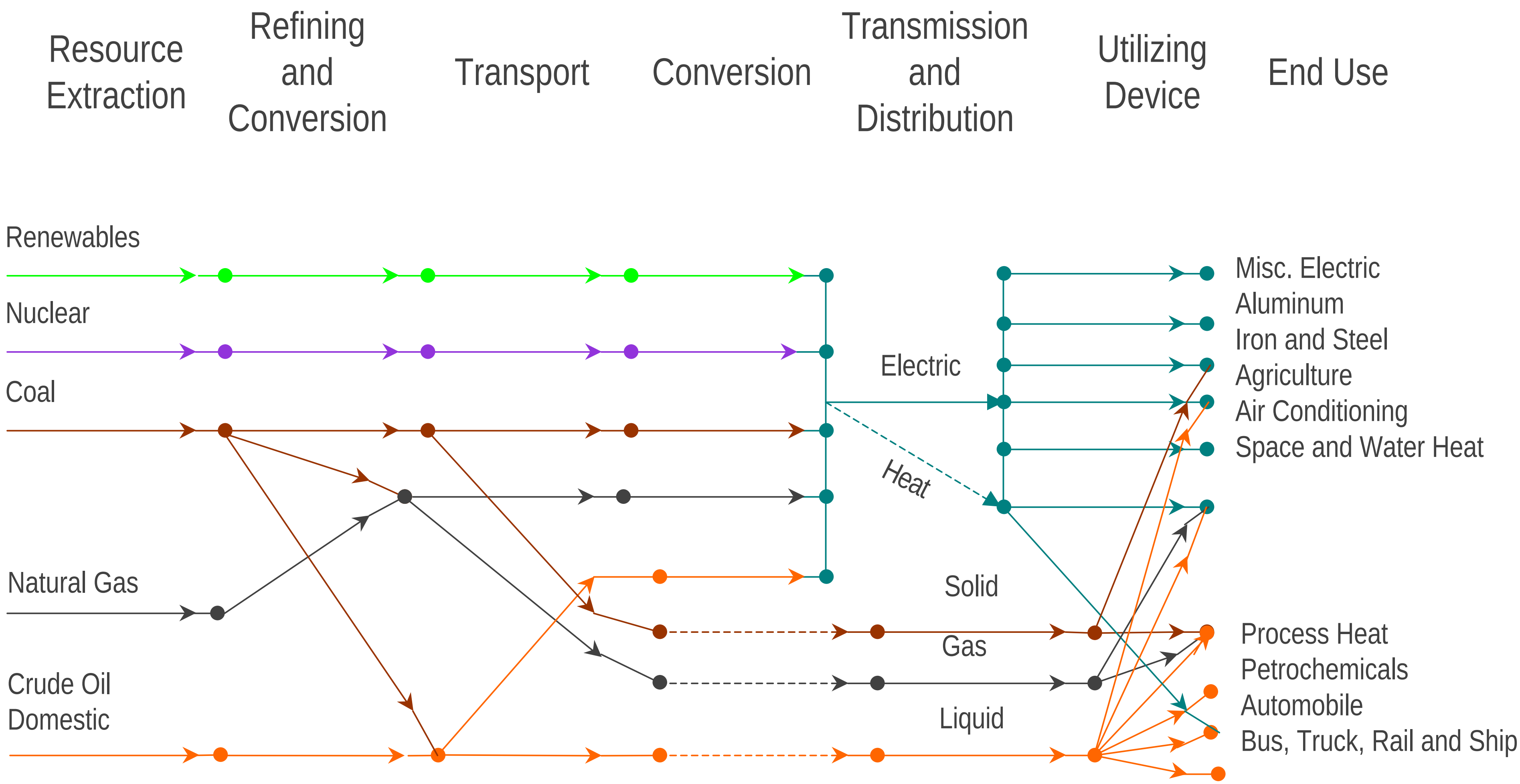
MARKAL is Used in Over 50 Countries



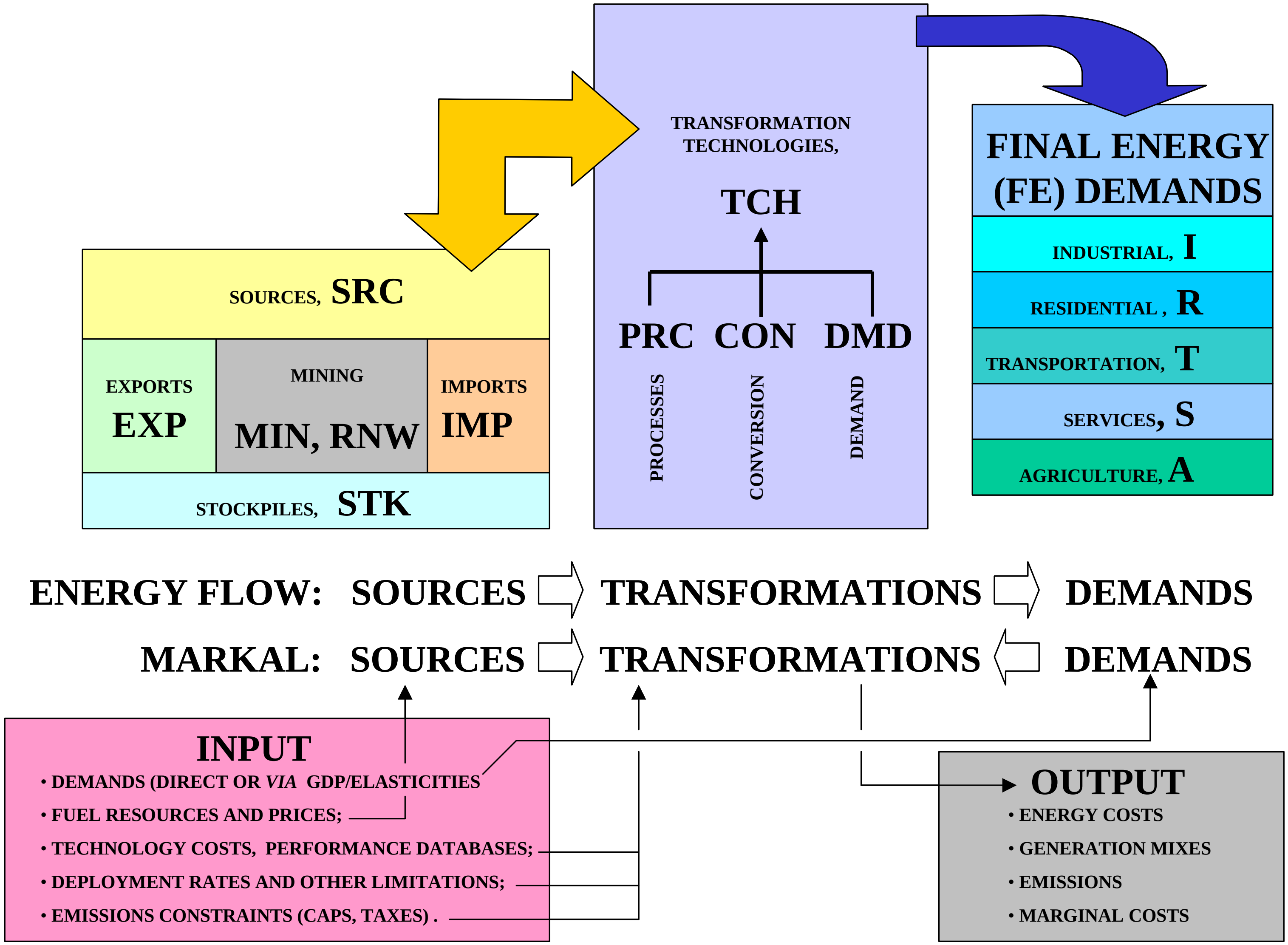
MARKAL Building Blocks



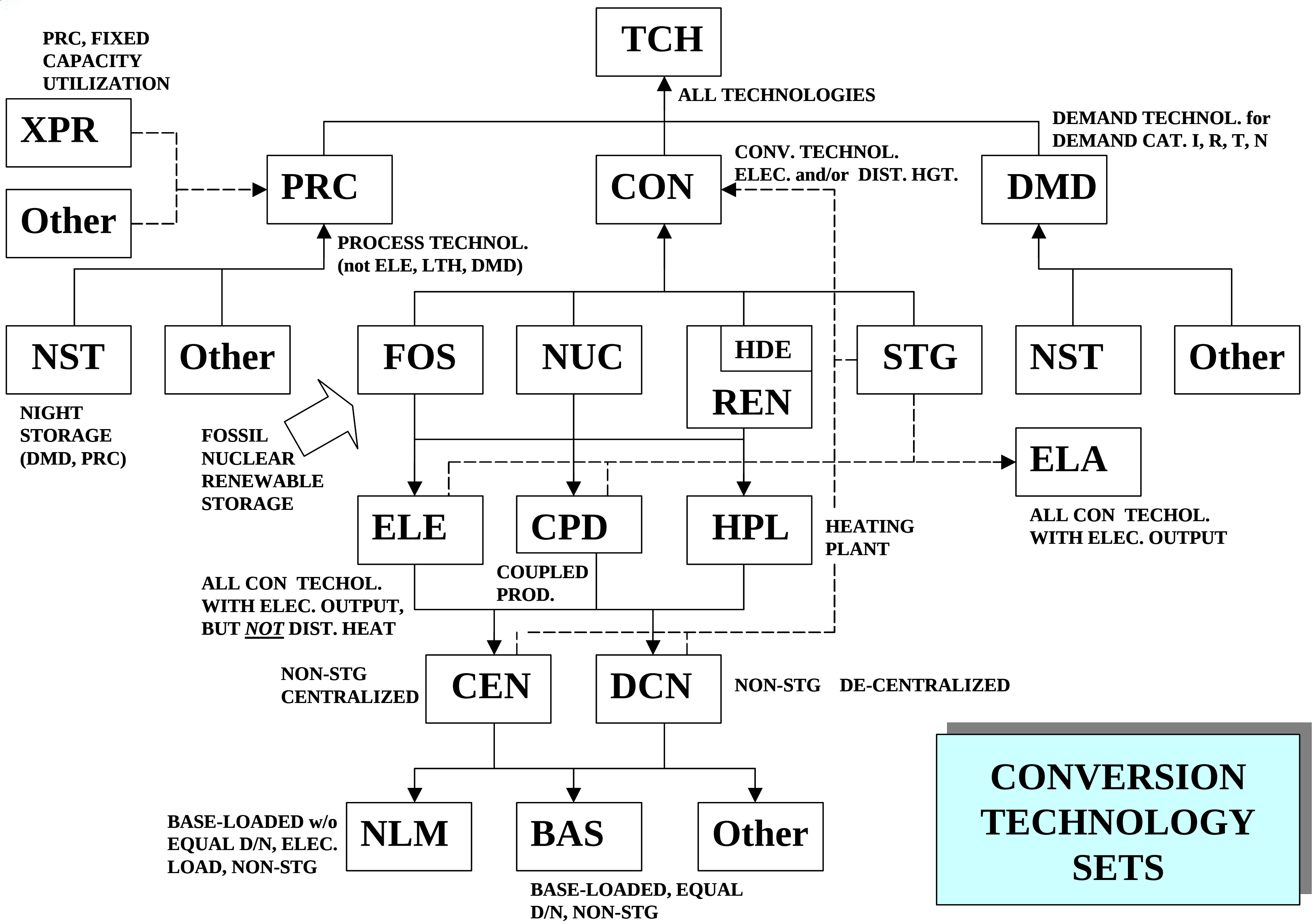
Simplified Reference Energy System (RES)



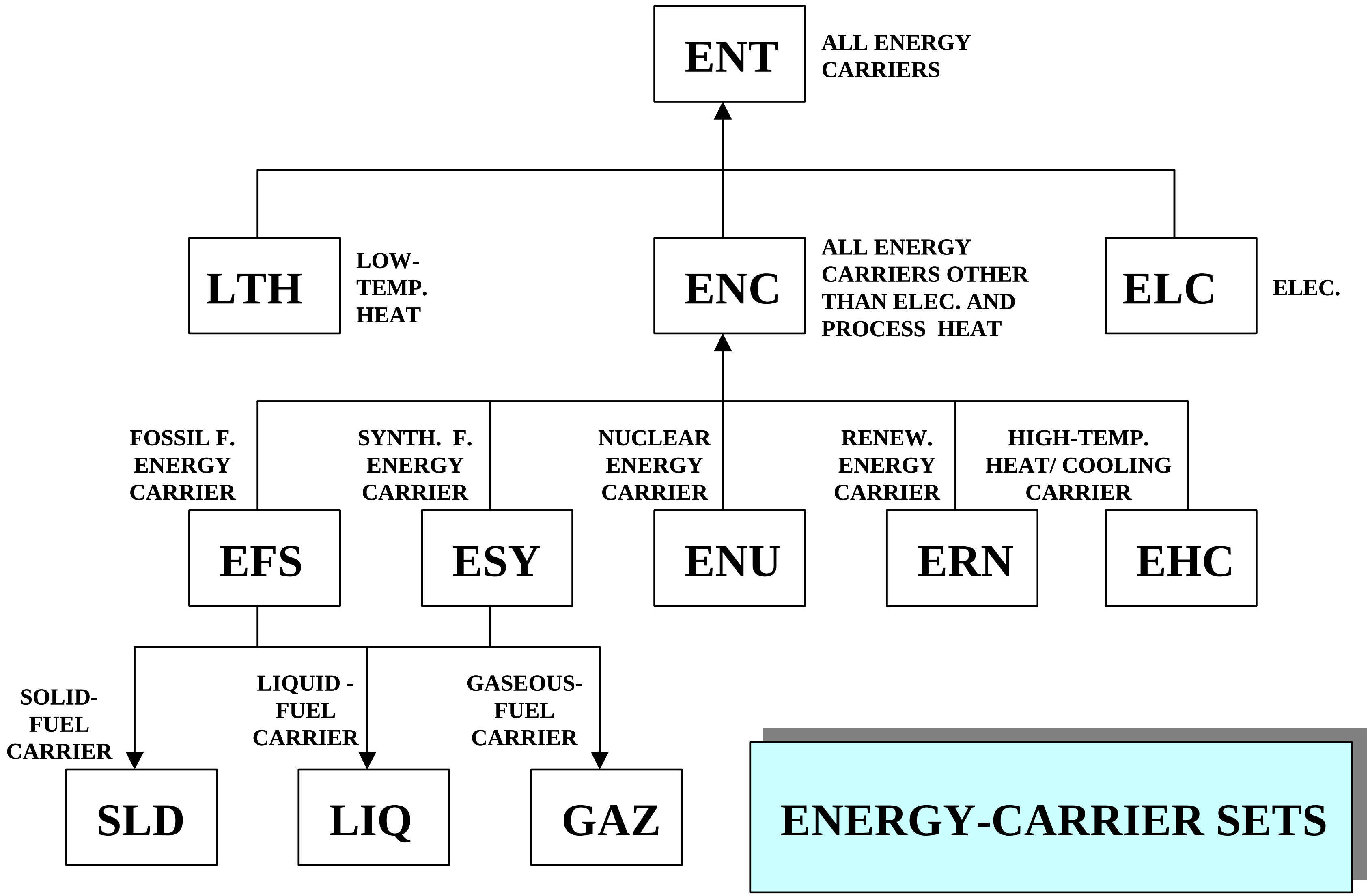
Energy Flows within MARKAL Showing Connectivity Between Sources {SRC}, Technologies {TCH}, and Demands {DMD}



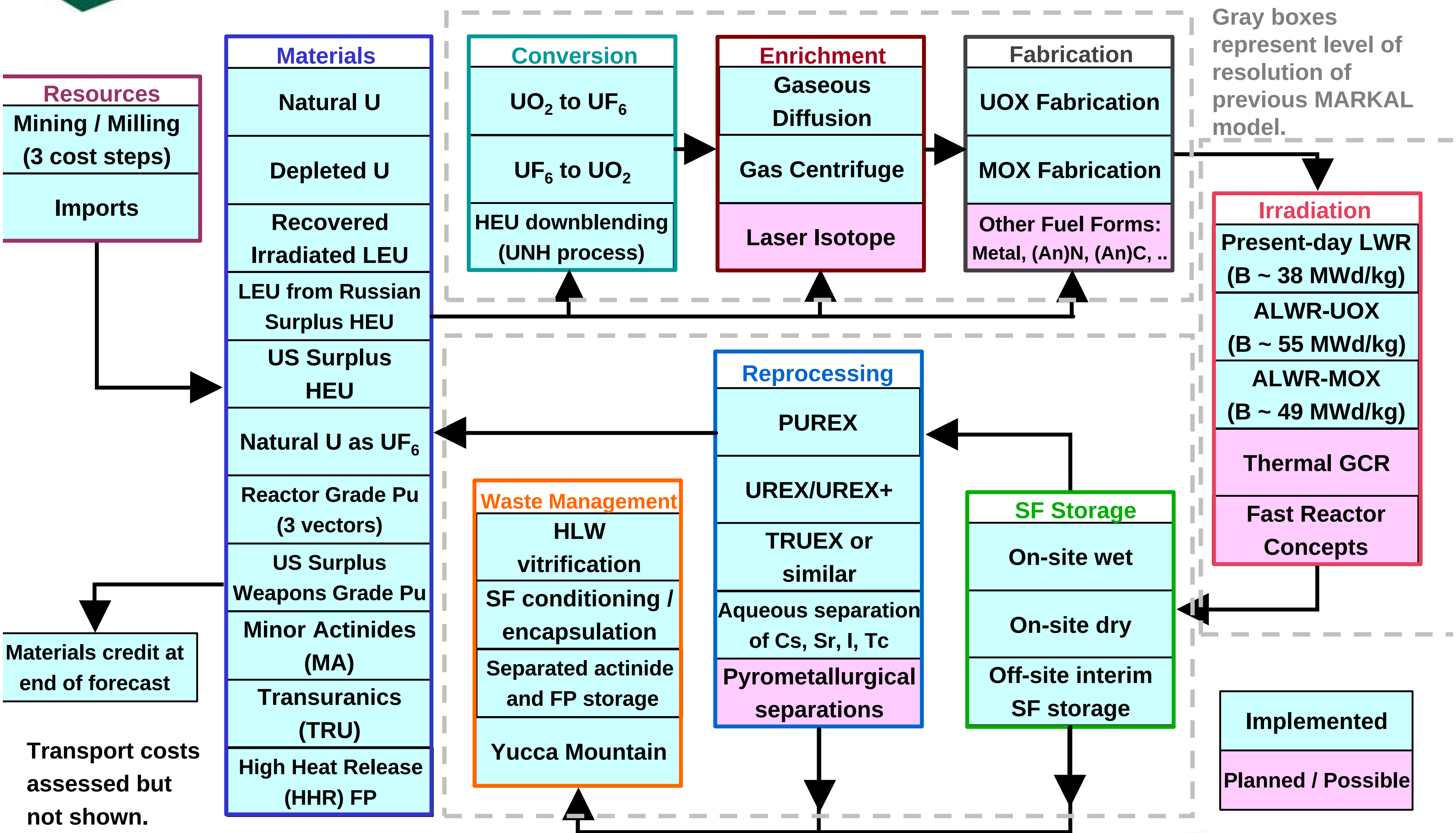
Connectivity Between Process {PRC}, Conversion {CON}, and Demand {DMD} Technologies {TCH} Modeled by MARKAL



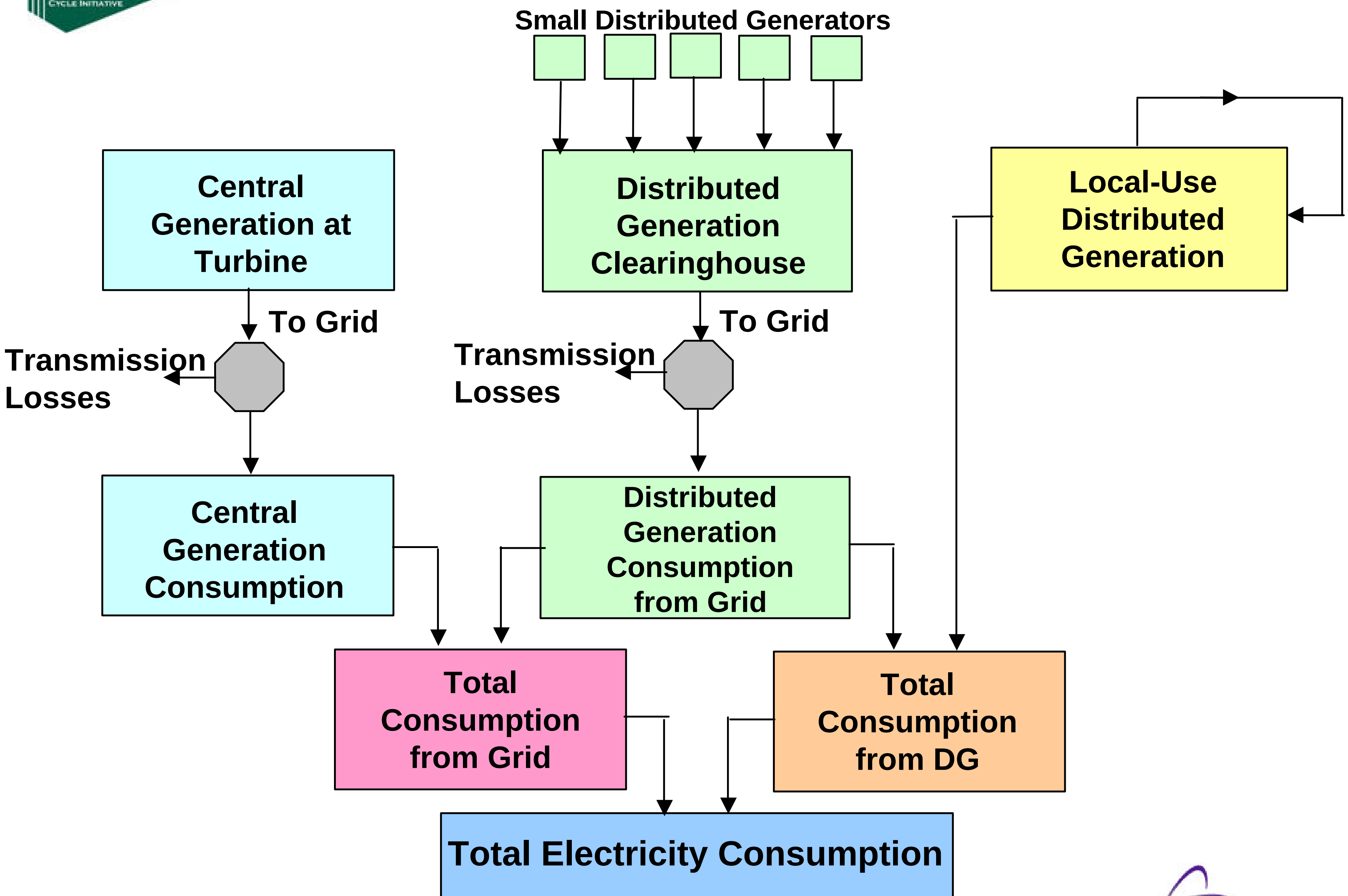
Connectivity Between Energy Carriers {ENT} Modeled by MARKAL



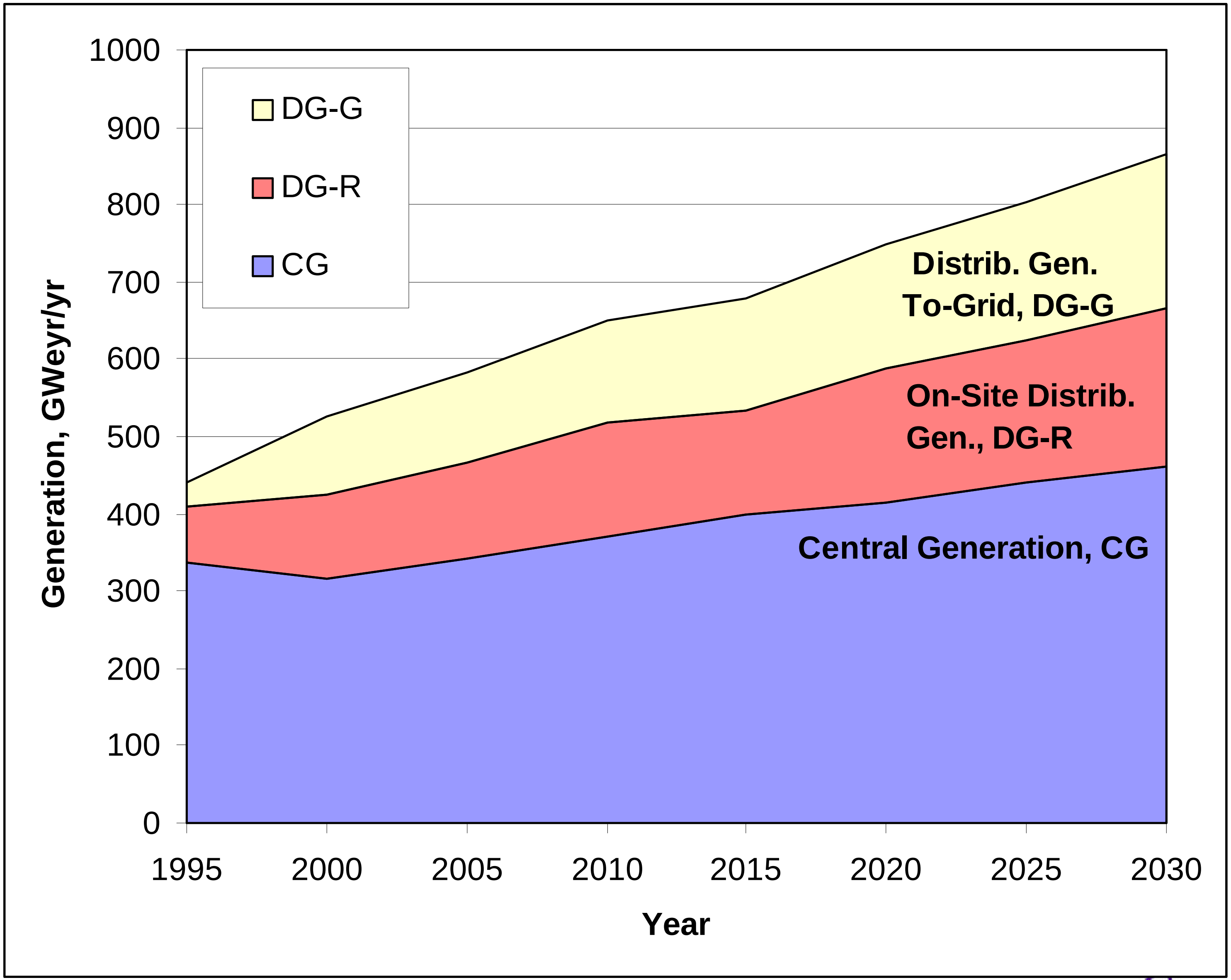
Nuclear Technologies and Materials Flows Implemented in (LA-)MARKAL Model



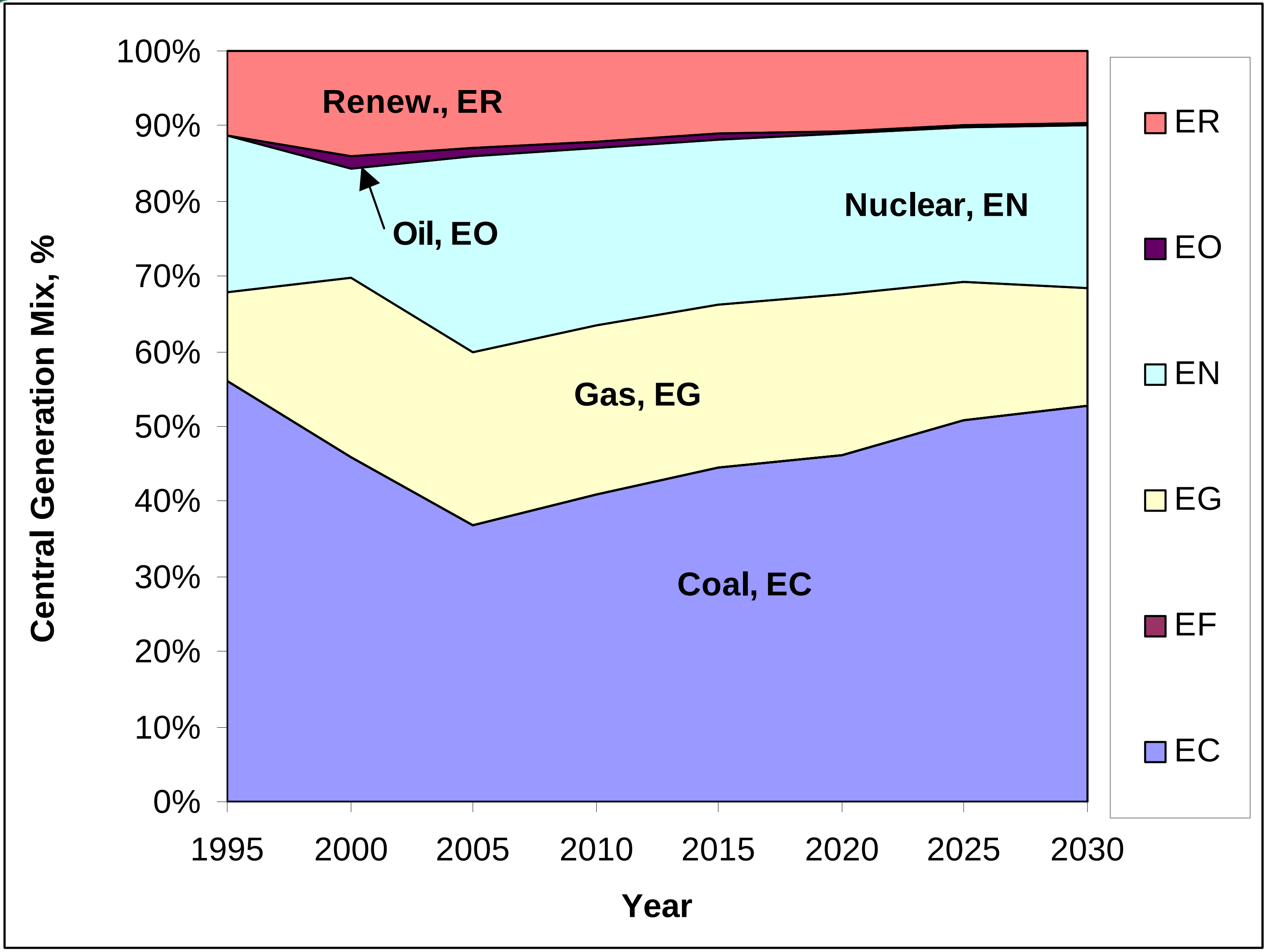
MARKAL Model for Distributed Electricity Generation (DG) *versus* Central Electricity Generation (CG)



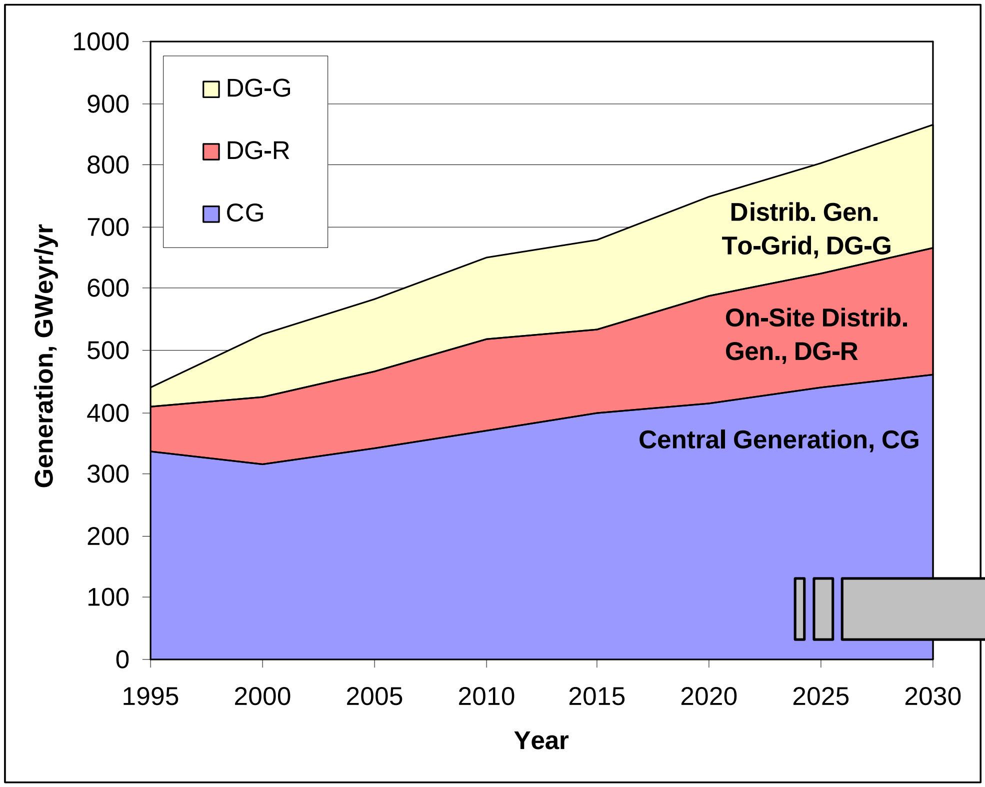
Preliminary MARKAL Results: Typical Mix Between Central and Distributed Generation

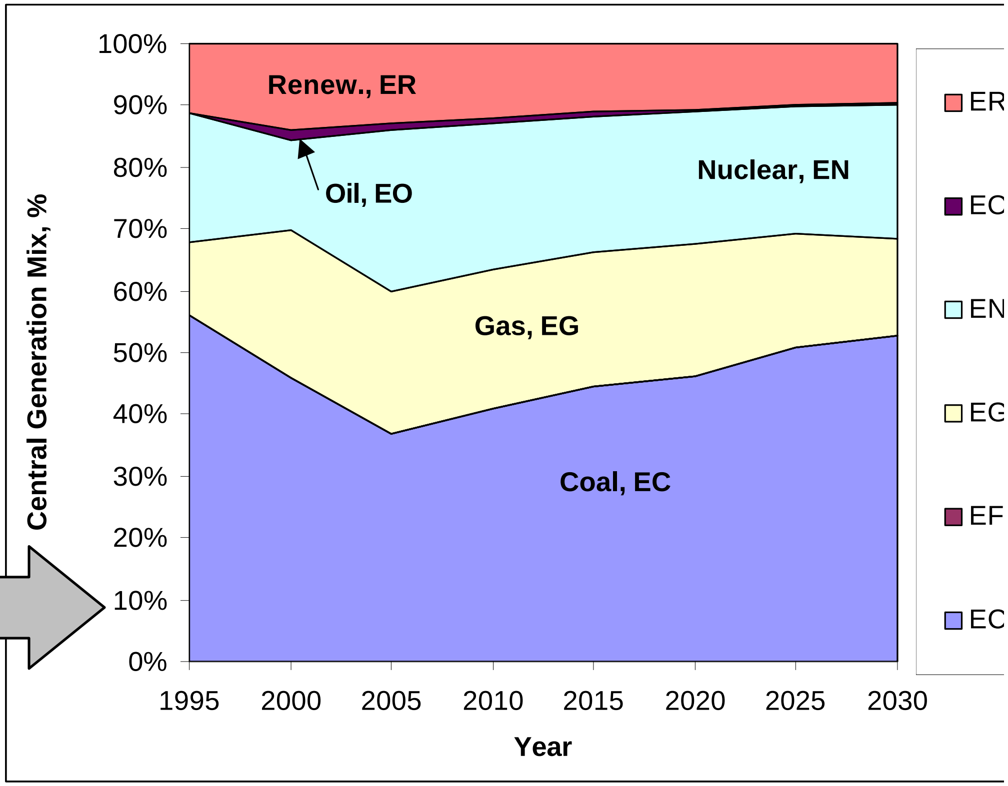


Preliminary MARKAL Result: Typical Mix for Central Generation



Preliminary MARKAL Results: Typical Mix Between Central and Distributed Generation





Neutronics - Modeling Support (High Pu/MA-recycle LWRs)

Focus of High-Recycle LWR Neutronics Studies is Placed on MIX[a] *rather than* CORAIL[b]

- **MIX *versus* CORAIL fuel assemblies:**

  - CORAIL: MOX is in outer fuel rods only; UO_2 is in inner fuel rods;

  - MIX: full cores of MOX fuel; MOX fuel is in all fuel rods; throughout each assembly; at least 12 fewer fuel rods per assembly (*versus* water holes) are required for safety;

- **MIX configuration can transmute legacy Pu, whereas CORAIL primarily deals with intrinsically generated Pu;**

- **MIX concept can be implemented in a specified number of LWRs so that in the future (*i.e.,* once legacy Pu is transmuted), all Pu generated from UO_2–fueled reactors can be transmuted in the MIX-fueled reactors.**

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[a] H. Trellue, "Reduction of the Radiotoxicity of Spent Nuclear Fuel Using a Two-Tiered System Comprised of Light Water Reactors and Accelerator-Driven Systems ", dissertation (to be published February, 2003);

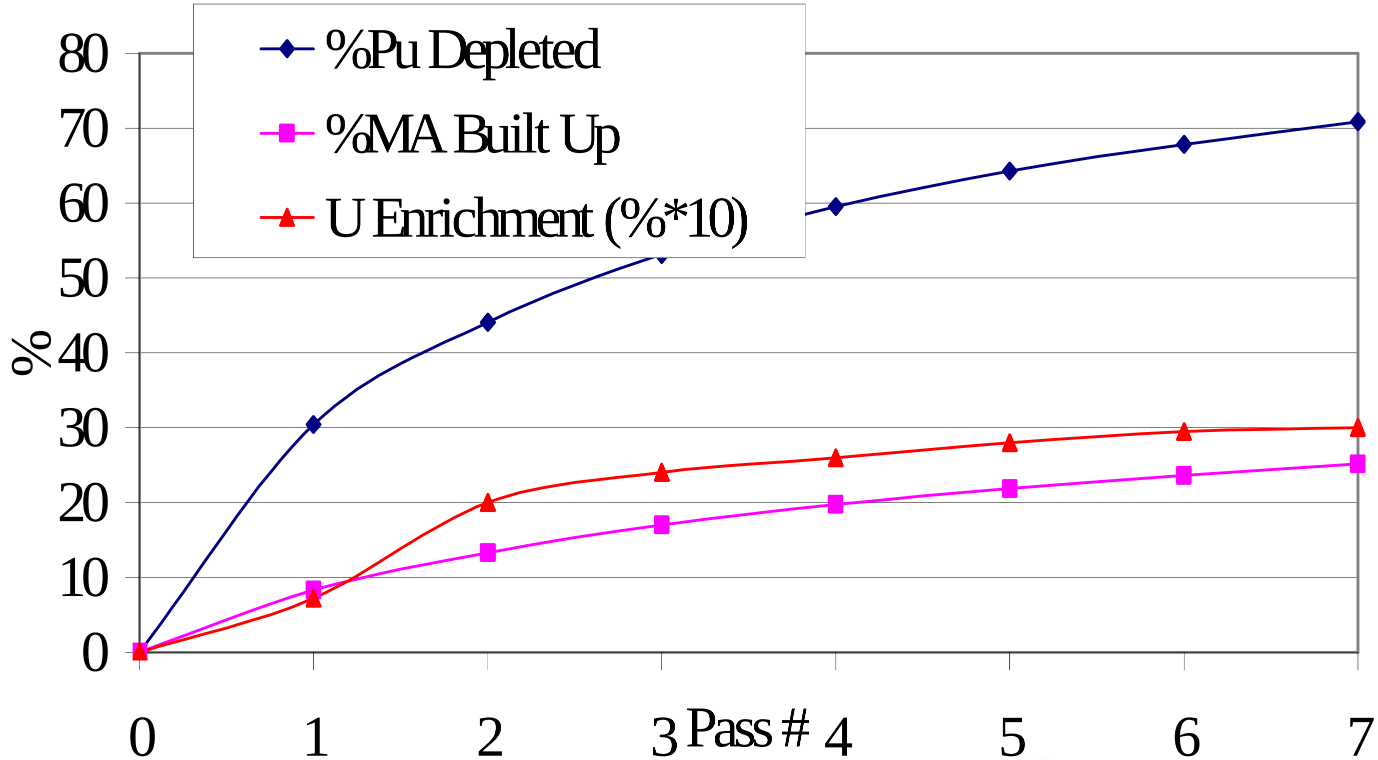
[b] G. Youinou, M. Delpech, J. L. Guillet, A. Puill, and S. Aneil, " Plutonium Management and Multi-Recycling in LWRs using an Enriched Uranium Support,"Global '99, August 29 – September 3, 1999 (Jackson Hole, Wy); T. K. Kim, J. A. Stillman, and T. A. Taiwo, "Assessment of TRU Stabilization in PWRs," Argonne National Laboratory document ANL-AAA-020 (August 14, 2002).

Neutronics Calculations for MIX Configuration

- Assume the use of full cores of MOX fuel;
- Only 10-20% of US LWRs fleet could transmute all US legacy Pu in ~50 years, as well as transmuting all Pu created by the remaining UO_2-fueled reactors under steady-state operation;
- Pu content in heavy metal in MOX is held at ~8.3 w%, and U enrichment in MOX is increased as a function of recycle to assure that criticality is maintained;
- Core parameters modified to meet neutronic safety constraints:
  - Twelve fuel rods replaced by water holes;
  - Soluble boron enrichment in water increased to ~25% ^{10}B;
  - Control rods changed to B_4C with up to 27.5% ^{10}B enrichment;
- Addition of minor actinides to MOX increases proliferation protection, but:
  - U enrichment was 2.7 w% for first pass, but had to be increased to 6.5 w/% for next passes:
  - 33.3% ^{10}B enrichment in control rods is required, even with the addition of four extra control rods.

Depletion of Plutonium in MIX Fuel Assemblies *versus* Number of Passes

- Pu that is burned each cycle is replaced with "fresh" Pu from SNF to help criticality and maintain a mass balance *per* reactor (*i.e.,* number of reactors required remains constant for each pass);
- Cooling time between cycles is 7 years (when activity and heat load of spent MOX fuel decreases to about that of extended burnup UO_2 fuel after 3 years);
- Starting with depleted U, the enrichment remains below limit of 5 w%;
- Three passes can transmute >50% Pu;
- About 1/3 of Pu transmuted is converted to minor actinides.



Neutronics-Based Proliferation Metrics

Four Proliferation-Relevant Attributes of Plutonium in a Multi-Recycling Nuclear Economy

AFC
Advanced Fuel Cycle Initiative

In addition to the quantity of Pu, the quality or weapons attractiveness of this material to a nuclear-weapon proliferant was examined.

Four proliferation-relevant attributes of the plutonium were quantified at three different junctures in the fuel cycle:

- Fissile Content [%];
- Heat Generation Rate [W/kgPu];
- Bare-Sphere Critical Mass [kgPu];
- Spontaneous Neutron Source [(n/s)/kgPu].

Proliferation risk reduction with increasing number of passes through the reactor is incremental rather than dramatic.

A Note on Proliferation Metrics Based on Pure Pu *versus* Pu with Minor Actinides

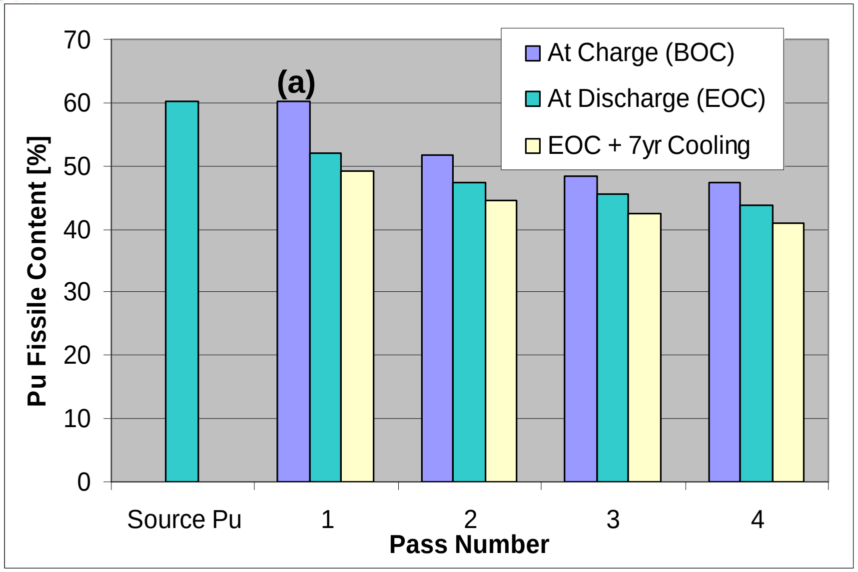
- **MA retention in MOX constitutes an additional hurdle for a proliferant, in that separated Pu is no longer directly available; Americium provides the bulk of the benefit (increased heat load);**

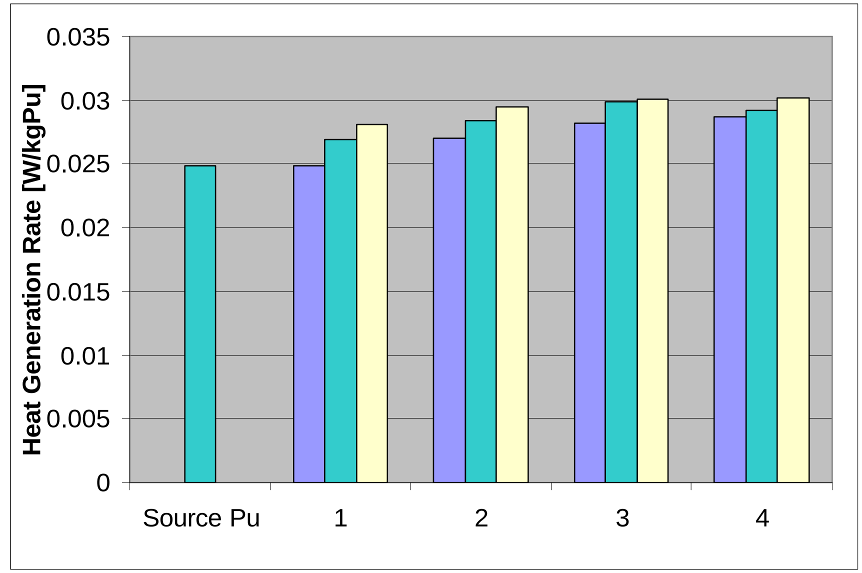
- **This barrier is porous, in that aqueous separation of plutonium is a mature, well-known technology. It may not be prudent to assume that proliferants do not possess an indigenous capability to separate plutonium from other actinides;**

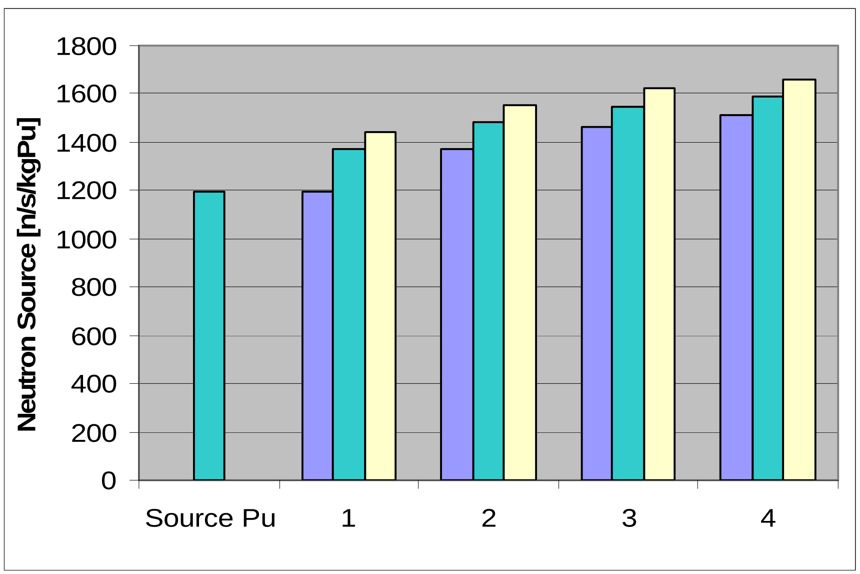
- **Therefore, regardless of the MA retention scenario, the evolution of the plutonium vector with recycle is of interest in assessing proliferation metrics;**

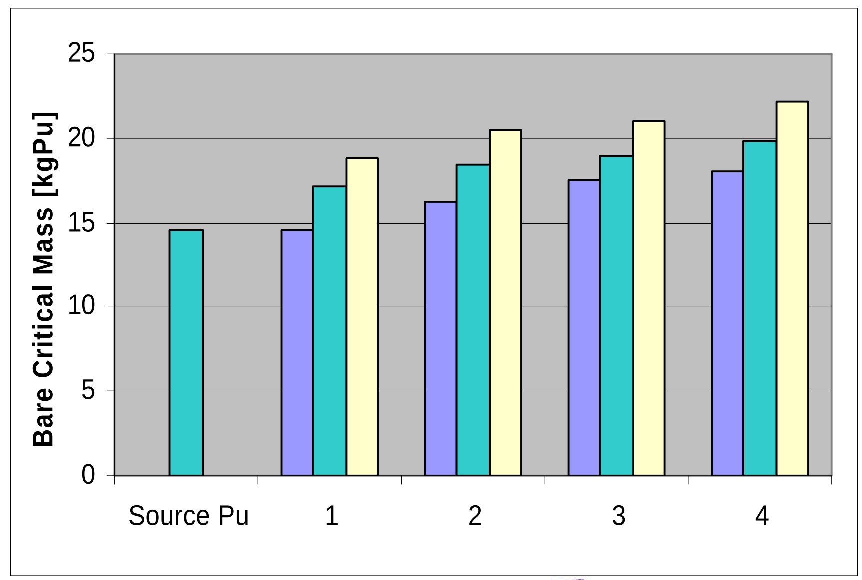
- **Small incremental improvements in the proliferation-relevant attributes of the plutonium vector as a function of recycle and MA retention scheme are seen; the bulk of this improvement follows from Np retention (*e.g.,* increase ^{238}Pu breeding).**

Multi-Recycled Plutonium Gradually Becomes Less Attractive to a NW Proliferant



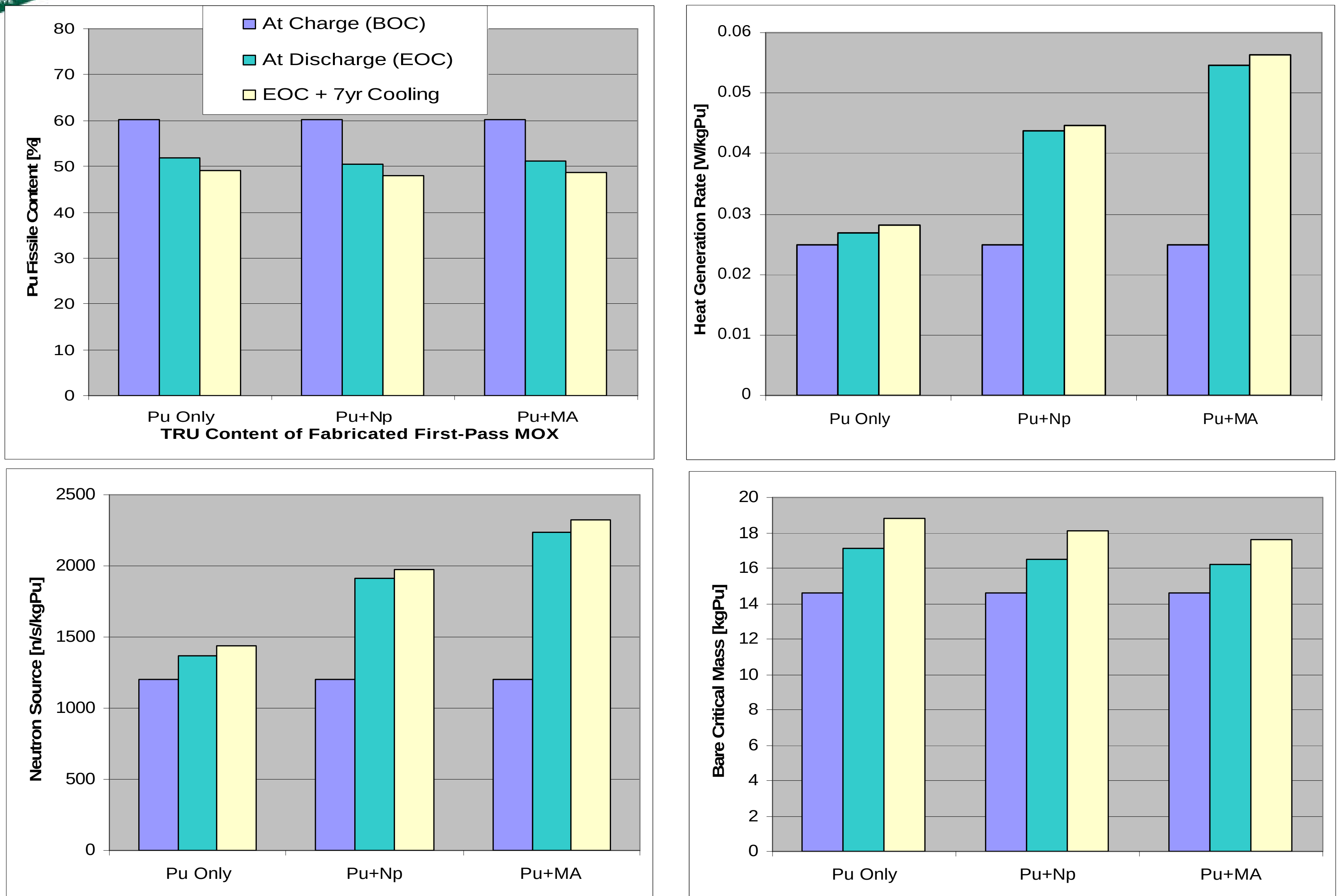




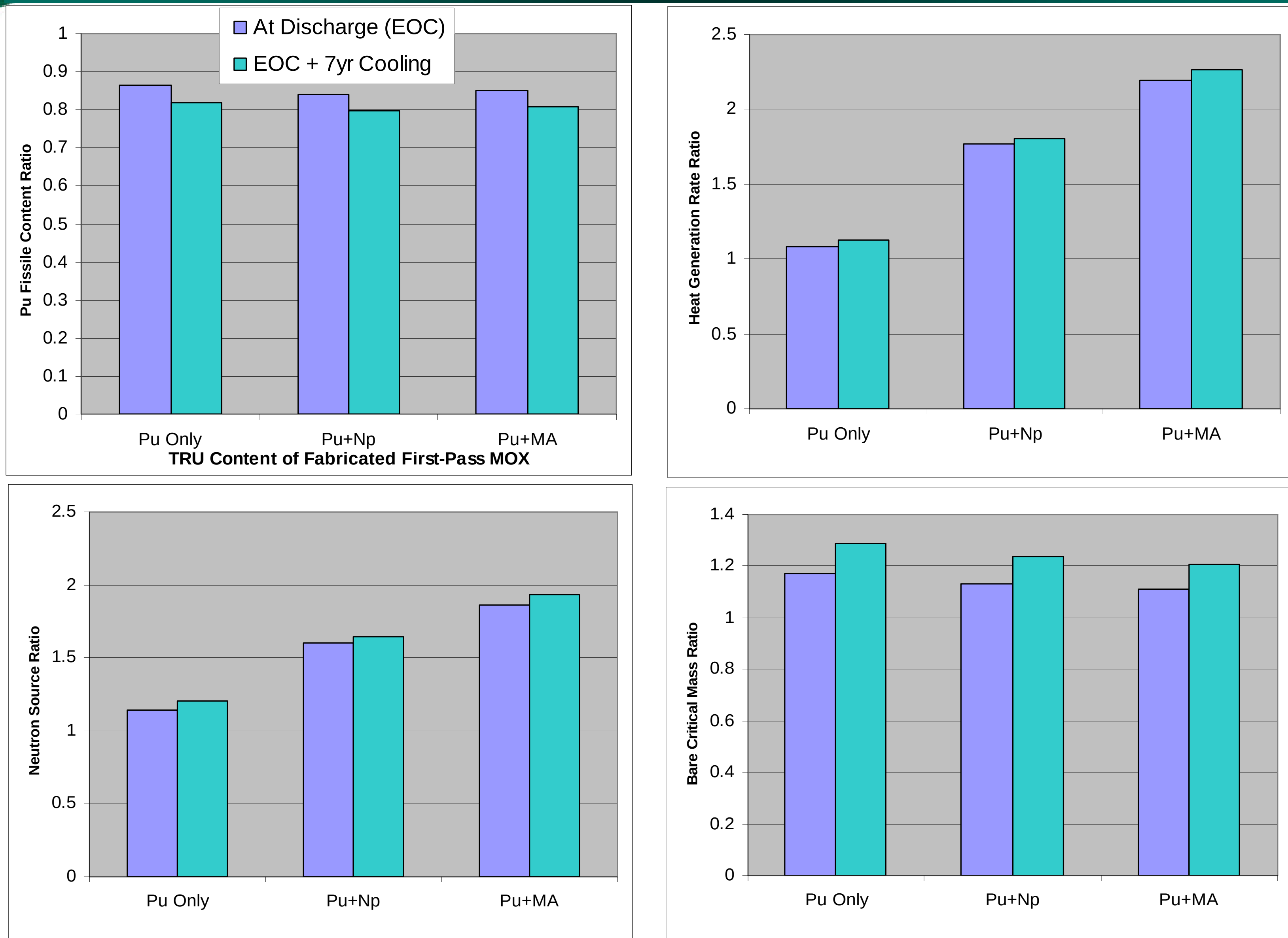


(a) Initial UOX-generated Pu constitutes the source in pass # 1

Proliferation-Relevant Features of Pu *versus* Composition of First-Pass MOX



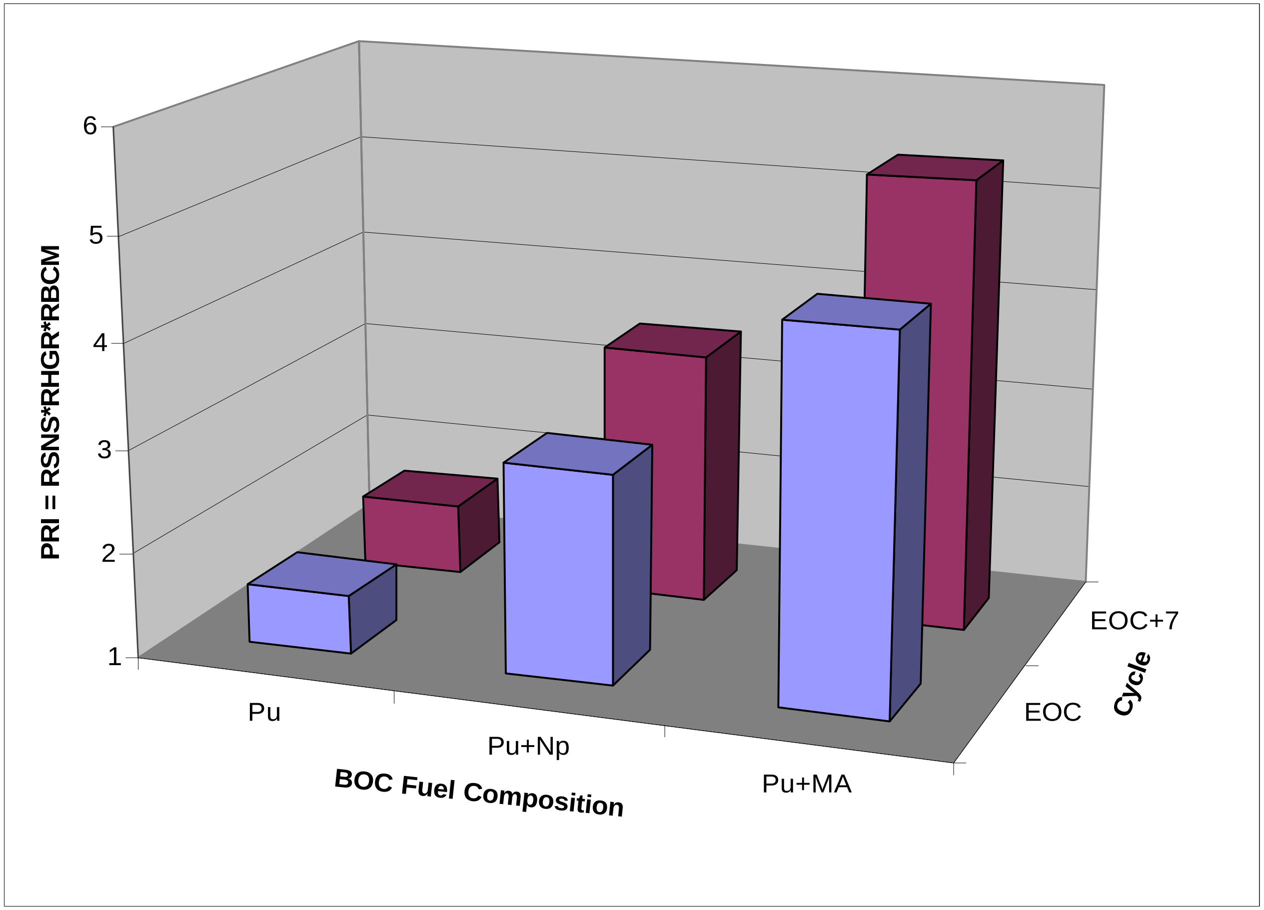
Proliferation Attributes of Plutonium in First-Pass MOX, Normalized to Reactor-Grade Pu (fresh SNF)



Relative Multi-Criteria Proliferation Risk Index

Measuring proliferation attractiveness in a way that recognizes undesirable characteristics of diverted Pu in a way that compounds the difficulty faced by a proliferant:

PRI = RSNS * RHGR * RBCM

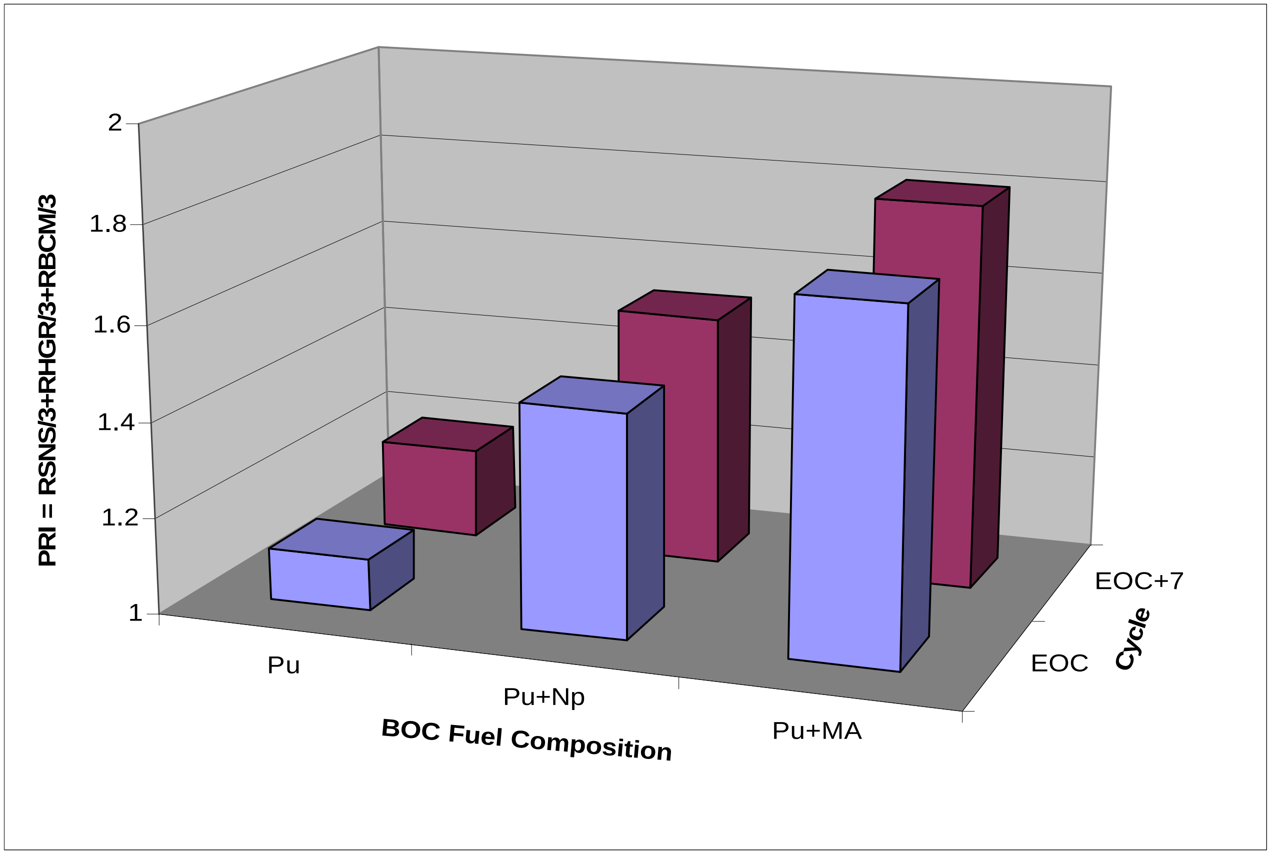


An Alternate Metric: Weighted, Additive Barriers to Proliferation

Another proliferation risk index wherein barriers are viewed as additive, reflecting a scenario in which obstacles to plutonium use are overcome independently:

$$PRI = w_{sns}*SNSR + w_{hgr}*HGRR + w_{bcm}*BCMR.$$

The weights w_i are chosen to sum to 1; for the case with equal barrier weighting (w_i = 1/3):

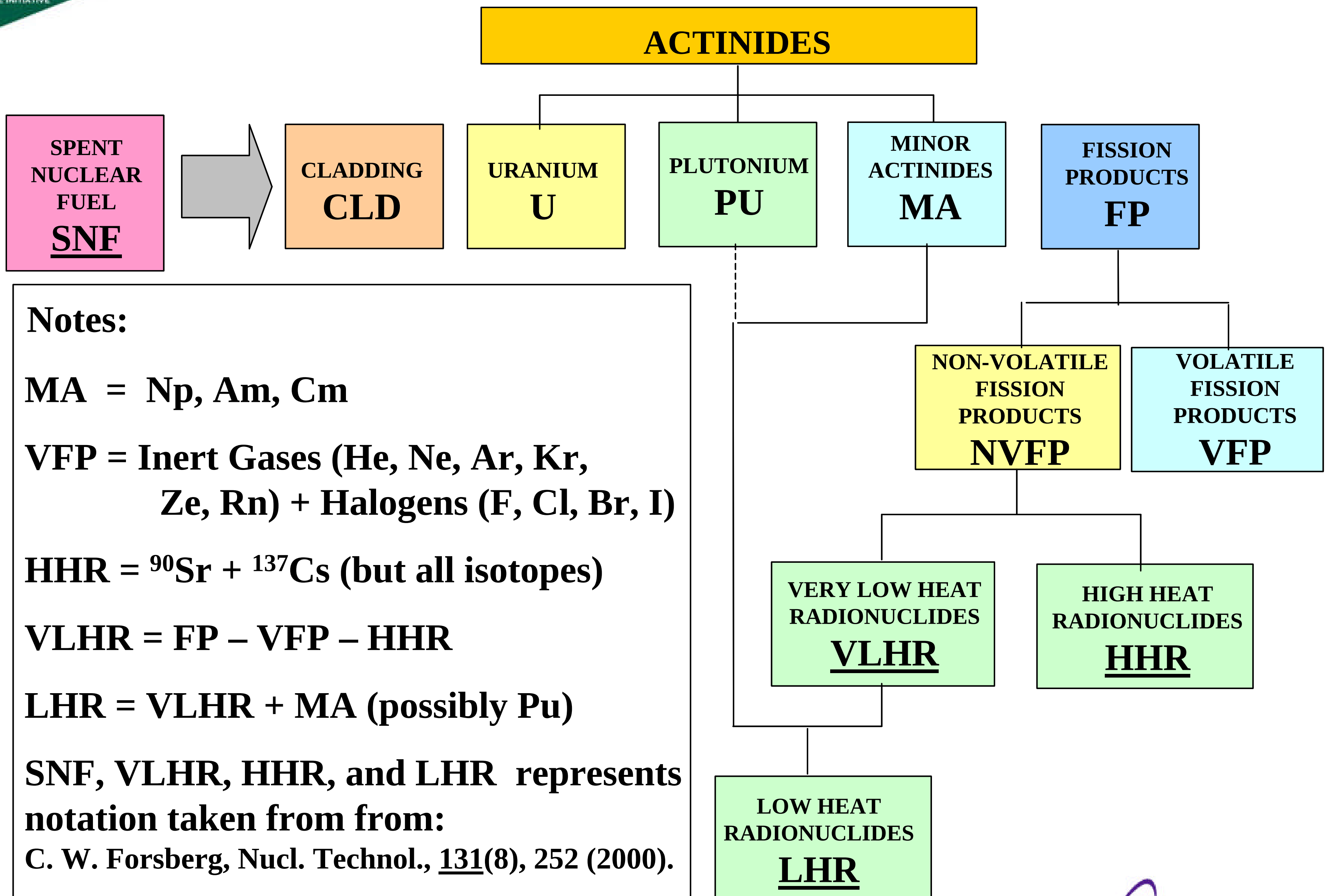


Yucca Mountain Business Model (YMBM)

Yucca Mountain Business Model (YMBM)

- Task: Evaluate effects of separations schemes on ***waste-carrying capacity*** of Yucca Mountain;
- Goal: to preclude or delay need for second repository;
- Capacity evaluated on basis of thermal effects on repository caused by radioactive decay heat:
  - Assume separated waste is vitrified before disposal (limited to 25 weight% waste in glass);
  - Repository temperature constraints include:
    - Waste package (vitrified waste) temperature - prevent crystallization of glass;
    - Tunnel wall temperature - prevent cracks that increase transport;
    - Far-field temperature - protect performance of zeolite layer below planned repository;
- Determine a "capacity increase" factor for a given separation scheme that is (approximately) independent of whether high-heat loading or low-heat loading repository scheme is used.

Front-end Repository Impacts: Key Components of Spent Nuclear Fuel As Related to Repository Thermal Impacts



Notes:

MA = Np, Am, Cm

VFP = Inert Gases (He, Ne, Ar, Kr, Ze, Rn) + Halogens (F, Cl, Br, I)

HHR = ^{90}Sr + ^{137}Cs (but all isotopes)

VLHR = FP – VFP – HHR

LHR = VLHR + MA (possibly Pu)

SNF, VLHR, HHR, and LHR represents notation taken from from:
C. W. Forsberg, Nucl. Technol., 131(8), 252 (2000).

Front-end Scenarios Adopted for Investigating Repository Impacts

Scenario, *nsc*	Short Description[c]	Elaborated Description[a]
1	Base Case	Direct disposal of SNF fuel assemblies
2	1 – U(ranium)	Vitrified [MA + Pu + NVFP]
3	2 – {Cs,Sr}	Vitrified [MA + Pu + VLHR = LHR][b]
4	3 - Pu	Vitrified [1 - U - HHR - Pu = MA + VLHR = LHR][b]
5	2 - Pu	Vitrified [1 - U - Pu = MA + NVFP][b]
7	4 - MA	Vitrified [1 - U – HHR - Pu – MA = VLHR][b]
6	5 - MA	Vitrified [1 - U - Pu - MA = NVFP][b]

[a] Disposed material form.

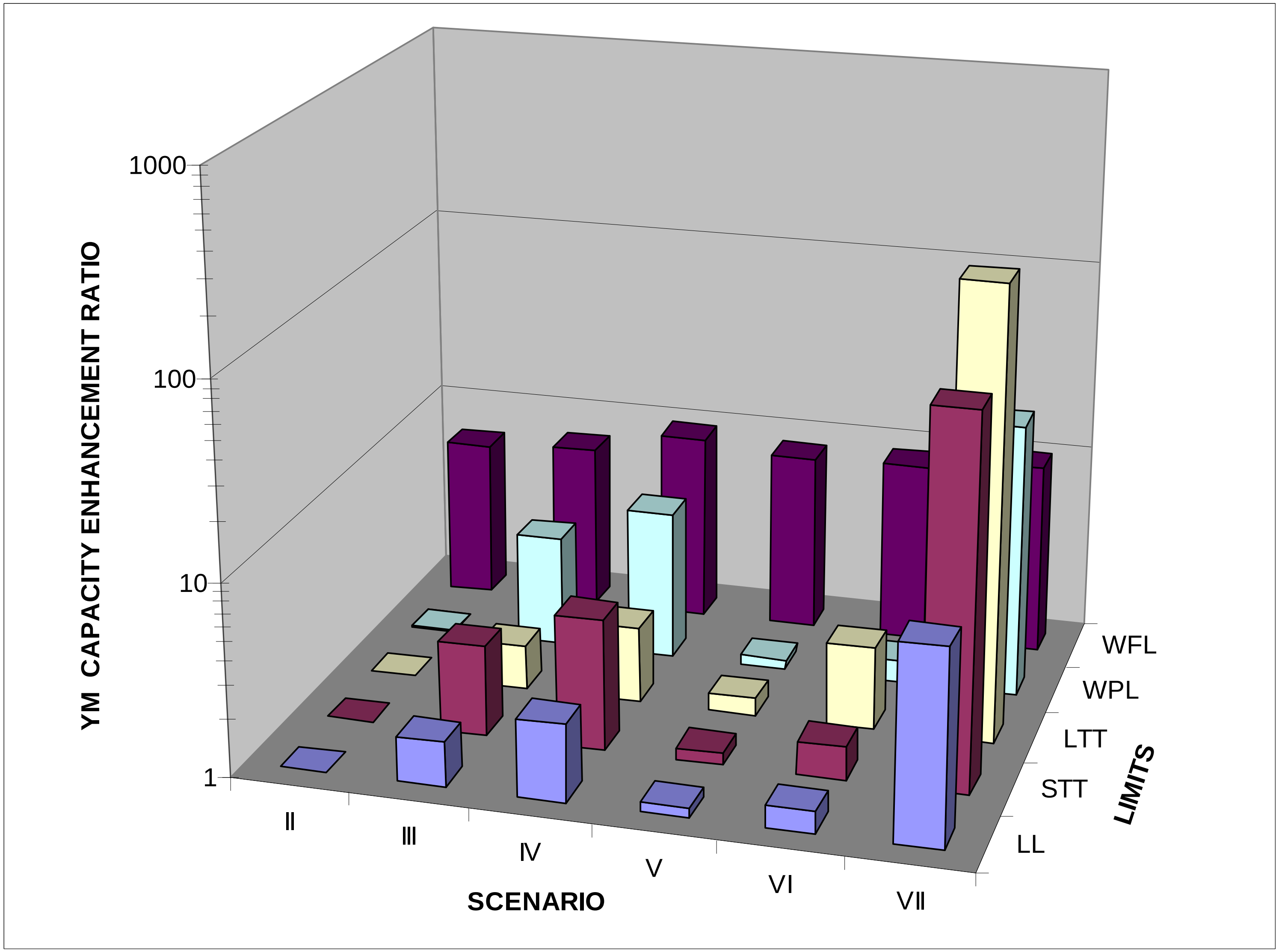
[b] MA = minor actinides; TRU = MA + Pu; NVFP = all non-volatile fission products; VLHR = very low heat radio-nuclides; LHR = low heat radio-nuclides; U = uranium. Note that *nsc = 3* and *4* result in two kinds of LHR waste products – with and without Pu; in a later paper (OECD, 2000) Forsberg includes Pu in the LHR mix.

[c] Expressed relative to the indicated scenario (*e.g., nsc = 2 = 1 – U* indicates scenario 1 with uranium removed via UREX process, and the remainder put into vitrified glass, *etc.*)

“Tier-less” Front-end Scenarios for Investigating Repository Impacts

Scenario, *nsc*	Elaborated Comparative Description
1	Base- or Point-of-Departure (POD) case: Direct disposal of SNF fuel assemblies, including most VFPs.
2	Reduce mass and (hopefully) volume, but must deali with full short- and long-term heat loads(b).
3	Reduce mass and (hopefully) volume, as well as short-term heat load associated with HHRs, but with full (TRU = Pu + MA) long-term heat load (and proliferation risk).
4	Similar to *nsc = 3*, with some reduction in long-term heat load through the removal of Pu (and reduced long-term proliferation risk).
5	Not unlike *nsc = 2*, but with some reduction in long-term heat load resulting from Pu removal (and reduced long-term proliferation risk).
6	Reduce mass and (hopefully) volume with full short-term heat load, but with significantly reduced long-term heat load.
7	The best it gets; volume and mass reduction along with reductions in both short-term and long-term heat loads.

Sample Result from YMBM: Capacity Enhancement Ratios *versus* Scenarios and Limiting Constraints[a]



LL = Least Limit;

STT = Short-Term Thermal (near-field tunnel wall, 40 yr);

LTT=Long-Term Thermal (far-field Zeolite, 300 yr);

WPL= Waste-Package Limit (center-line temperature);

WFL = Waste-Fraction Limit (glass loading limit).

Interim Results from YMBM Emplacement Studies

- Removing high heat-load species greatly increased repository capacity, limited by waste form (*e.g.,* by waste content in vitrified waste) to ~ 9-fold increase;
- Removal of both short-lived (Cs, Sr, and their decay products) and long-lived (actinides) species is necessary to achieve significant repository capacity increases;
- Increased repository capacity requires separate disposal (transmutation) of actinides (Pu, Am, Cm) plus alternate disposal (engineered storage or short-term repository) of short-lived species;
- Separate waste streams (short-lived fission products, hulls and clad, vitrified waste) are generated, but these added waste streams may be handled at reduced cost (to be evaluated later in FY 2003).

Addition of Disposal Costing Model Based Upon Repository Heat-Load Limitations (Interim)

- **Unit repository disposal costs for spent fuel, less transportation-related charges, are currently estimated by OMB as *~$440/kgIHM;***

- **Question: How would disposal costs that include vitrification as well as emplacement compare if a reprocessing / HLW vitrification strategy were pursued?**

- **A preliminary methodology for evaluating these costs is proposed that uses guidelines[a] based on heat-release limitation .**

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[a] **BATHKE, C.G. *et. al*., "Advanced Nuclear Fuel Cycle Systems Analyses for FY 2002," Los Alamos National Laboratory document LA-UR-02-6674 (October 25, 2002).**

Repository Utilization as a Function of Waste Content and HLW Composition[a]

	Waste composition	Waste mass [kg/kgIHM]	Repository utilization [kg/kgIHM]	HLW packing density [kg/m^3 glass]
I[b]	All SNF	1.00	1.00	N/A
II	TRU, all FP	0.0516	1.00	92.8
III	TRU, LHRFP	0.0475	0.585	323
IV	MA, LHRFP	0.0380	0.398	396
V	MA, all FP	0.0420	0.893	81.4
VI	LHRFP	0.0366	0.099	625
VII	All FP	0.0407	0.769	90.3

[a]For a standard burnup LWR with *~10-year* cooling prior to reprocessing and disposal;

[b]Direct disposal SNF is included for comparison.

Interim Conclusion for Disposal Cost as a Function of Waste Content

Using heat load as the sole YM design criterion, the disposal cost may be formulated based on:

- **HLW unit vitrification cost of *300,000 $/m^3* [a];**

- **HLW unit repository disposal cost of *332 $/kgSNF(eq.)* of YM capacity used;**

This condition represents the *$440/kg* LCC estimate minus the (avoided) YM cost component relating to spent fuel waste package fabrication.

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[a] **Hanford HLW vitrification program, "High-Level Waste Melter Study Report", Pacific Northwest National Laboratory report PNNL-13582 (July, 2001).**

Disposal Cost Comparison Made Under YMBM "Rules" (*e.g.*, partial costing)

	Waste originating from 1 kgHM [kg waste]	Unit vitrification cost [$/kg waste]	Unit emplacement/ disposal cost [$/kg waste]	Total [conditioning + disposal] [$/kgIHM]	'Effective' repository capacity [tonneIHM]
I	1.00	N/A	440	440	83,800[a]
II	0.0516	3231	6,436	498	83,800
III	0.0475	922	4,087	238	143,300
IV	0.0380	757	3,484	161	210,300
V	0.0420	3,686	7,052	451	93,900
VI	0.0366	480	897	50	846,400
VII	0.0407	3,323	6,274	390	108,900

[a] DOE Office of Civilian Radioactive Waste Management design basis; "Analysis of the Total System Life Cycle Cost of the Civilian Radioactive Waste Management Program,", US Department of Energy report DOE/RW-0533, (2001).

CEA/USDOE (ANL, LANL) Collaboration

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- Conduct CEA-LANL/ANL dynamic NFC model benchmarking (COSI-NFCSim):

  - align NFCSim and COSI neutronics, materials balance, costing, *etc.* processing capabilities;

  - NFC benchmarking scenarios (open cycle, single Pu recycle in ALWRs commencing in 2015, Pu+Np recycle) to be finalized by 02/07/2003, results available for comparison as of 03/15/2003;

- CEA/DOE joint reference scenarios study, to commence thereafter:

  - LWR + ALWR (beginning 2015) with Pu or TRU recycle;

  - LWR + ALWR + FR with recycle (beginning 2030).

Top-Level Summary of Parameters to be Determined and Agreed Between COSI and NFCSim Simulation Models

- **Growth rate of nuclear-energy demand;**
- **Number of recycles (LWR, ALWR, FR);**
- **What is recycled (carried over; LWR, ALWR, FR);**
- **FR conversion ratio;**
- **Reactor parameter matrices (efficiency, availability, burnup, *etc*.);**
- **Cost and financial parameters (unit costs, fixed and variable O&M, interest rates, tax structure, debt-to-equity);**
- **Time database (cooling time, processing lags, transportation, construction, R&D/technology lags);**
- **Material loss fractions in fabrication, processing, *etc*.**
- **Separation and disposal (S&D) strategies.**